

DUDLEY KNOX LIBRARY
NAVAL POSTGRADUATE SCHOOL
MONTEREY, CALIF 93940

NAVAL POSTGRADUATE SCHOOL

Monterey, California



THESIS

AN INTERACTIVE CODE FOR A PRESSURIZED WATER REACTOR
INCORPORATING TEMPERATURE AND XENON FEEDBACK

by

Gregory Garver Heath

June 1980

Thesis Advisor:

P. J. Marto

Approved for public release; distribution unlimited

T196581

REPORT DOCUMENTATION PAGE		READ INSTRUCTIONS BEFORE COMPLETING FORM
1. REPORT NUMBER	2. GOVT ACCESSION NO.	3. RECIPIENT'S CATALOG NUMBER
4. TITLE (and Subtitle) An Interactive Code for a Pressurized Water Reactor Incorporating Temperature and Xenon Feedback		5. TYPE OF REPORT & PERIOD COVERED Master's Thesis: June 1980
		6. PERFORMING ORG. REPORT NUMBER
7. AUTHOR(s) Gregory Garver Heath		8. CONTRACT OR GRANT NUMBER(s)
9. PERFORMING ORGANIZATION NAME AND ADDRESS Naval Postgraduate School Monterey, California 93940		10. PROGRAM ELEMENT, PROJECT, TASK AREA & WORK UNIT NUMBERS
11. CONTROLLING OFFICE NAME AND ADDRESS Naval Postgraduate School Monterey, California 93940		12. REPORT DATE June 1980
		13. NUMBER OF PAGES 93
14. MONITORING AGENCY NAME & ADDRESS (if different from Controlling Office) Naval Postgraduate School Monterey, California 93940		15. SECURITY CLASS. (of this report) Unclassified
		16. DECLASSIFICATION/DOWNGRADING SCHEDULE
16. DISTRIBUTION STATEMENT (of this Report) Approved for public release; distribution unlimited		
17. DISTRIBUTION STATEMENT (of the abstract entered in Block 20, if different from Report)		
18. SUPPLEMENTARY NOTES		
19. KEY WORDS (Continue on reverse side if necessary and identify by block number) Pressurized Water Reactor, Xenon Feedback, Moderator Feedback		
20. ABSTRACT (Continue on reverse side if necessary and identify by block number) An interactive computer model of a highly enriched pressurized water reactor was developed, using the applicable plant parameters from Shippingport Atomic Power Station. The point reactor kinetics equations for one delayed neutron precursor group were linearized using small perturbation theory. The model included both moderator and Xenon-135 reactivity feedback effects, as well as an automatic reactor protection and average reactor coolant temperature control system. The thermal response of the model plant was simulated for normal operating transients induced either by control rod or turbine load changes. The		

post-shutdown Xenon transient response was also modeled. The interactive program was coded in FORTRAN-IV language, and the simulation program was coded in IBM CSMP-III language.

Approved for public release; distribution unlimited

An Interactive Code for a
Pressurized Water Reactor
Incorporating
Temperature and Xenon Feedback

by

Gregory Garver Heath
Lieutenant Commander, United States Navy
B.S.M.E., United States Naval Academy, 1971

Submitted in partial fulfillment of the
requirements for the degree of

MASTER OF SCIENCE IN ENGINEERING SCIENCE

from the

NAVAL POSTGRADUATE SCHOOL
June, 1980

ABSTRACT

An interactive computer model of a highly enriched pressurized water reactor was developed, using the applicable plant parameters from the Shippingport Atomic Power Station. The point reactor kinetics equations for one delayed neutron precursor group were linearized using small perturbation theory. The model included both moderator and Xenon-135 reactivity feedback effects, as well as an automatic reactor protection and average reactor coolant temperature control system. The thermal response of the model plant was simulated for normal operating transients induced either by control rod or turbine load changes. The post shutdown Xenon transient response was also modeled. The interactive program was coded in FORTRAN-IV language, and the simulation program was coded in IBM CSMP-III language.

TABLE OF CONTENTS

I. INTRODUCTION -----	9
II. MODEL DESIGN CONSIDERATIONS -----	11
A. PRESSURIZED WATER REACTOR -----	11
B. MODEL AND PROGRAM FLEXIBILITY -----	13
III. MODEL DEVELOPMENT -----	15
A. POINT REACTOR KINETICS EQUATIONS -----	15
1. Zero Power Reactor Transfer Function -----	17
2. Limitations -----	18
B. REACTIVITY FEEDBACK MECHANISMS -----	19
1. Xenon-135 Transfer Function -----	21
2. Moderator Temperature Feedback -----	24
C. THERMAL ANALYSIS -----	26
1. Reactor Heat Transfer Function -----	26
2. Heat Exchanger Transfer Function -----	29
3. Primary Piping Transfer Functions -----	35
D. OVERALL PLANT BLOCK DIAGRAM -----	37
E. MODEL CONSTANTS -----	38
IV. RESULTS -----	42
A. INTERACTIVE PROGRAM -- -----	42
B. SIMULATION PROGRAM---- -----	48
V. CONCLUSIONS AND RECOMMENDATIONS -----	54
APPENDIX A: DEVELOPMENT OF THE XENON FEEDBACK TRANSFER FUNCTION COMPONENTS -----	56
APPENDIX B: DEVELOPMENT OF THE REACTOR HEAT TRANSFER FUNCTION -----	61

APPENDIX C: DEVELOPMENT OF THE HEAT EXCHANGER TRANSFER FUNCTION -----	65
APPENDIX D: DEVELOPMENT OF THE PRIMARY PIPING FUNCTIONS -----	67
APPENDIX E: INTERACTIVE PROGRAM LISTING -----	71
APPENDIX F: SIMULATION PROGRAM LISTING -----	78
APPENDIX G: INTERFACE PROGRAM LISTING -----	91
LIST OF REFERENCES -----	92
INITIAL DISTRIBUTION LIST -----	93

LIST OF FIGURES

1. Schematic of the Model Plant -----	12
2. Block Diagram of the Zero Power Reactor Transfer Function -----	18
3. Block Diagram of the Reactor Transfer Function with Xenon-135 and Moderator Temperature Feedback Loops, and External Reactivity -----	20
4. Xe-135 Decay Scheme -----	21
5. Simplified Xenon-135 Decay Scheme -----	22
6. Block Diagram of the Reactor Transfer Function with Xenon-135 Feedback and External Reactivity -----	24
7. Block Diagram of the Reactor Heat Transfer Function -----	29
8. Block Diagram of the Heat Exchanger Transfer Function -----	34
9. Block Diagram of the Transport and Mixing Delay Functions -----	38
10. Overall Plant Block Diagram -----	39
11. Interactive Program, Introduction and Instructions -----	43
12. Interactive Program, Initial Power Level and Simulation Choice Inputs -----	44
13. Post-shutdown Xenon-135 Behavior -----	45
14. Interactive Program, Control Rod Movement Inputs -----	46
15. Interactive Program, Turbine Load Change Inputs -----	47
16. Interactive Program, Logic Routine Example -----	49
17. Power, Reactivity, and Temperature Response to a 20 to 40% Turbine Load Change -----	50
18. Power, Reactivity, and Temperature Response to a 60 to 40% Turbine Load Change -----	51
19. Power, Reactivity, and Temperature Response to a 10 second Inward Control Rod Movement -----	52

20.	Power, Reactivity, and Temperature Response to a 10 second Outward Control Rod Movement -----	53
D-1.	Mixing Volume -----	69

I. INTRODUCTION

Computers have been used in the design and analysis of nuclear reactors since the inception of reactor technology, and many codes have been developed. More recently, digital computer programs have been used as learning devices for nuclear engineering students. The formulation of the computational problem of predicting the behavior of a nuclear reactor has been greatly facilitated with the use of high level computer languages.

The purpose of this work was to develop a computer assisted learning device to be used by students taking nuclear engineering courses at the Naval Postgraduate School. The program graphically displays the simulated kinetic and thermal transient responses of a pressurized water reactor power plant. The reactivity feedback effects of Xenon-135 poisoning and moderator temperature are incorporated into the model. Normal operating transients, starting from a steady state critical condition, can be induced by ramp changes in either control rod position or turbine load. The initial reactor power level and reactivity change mechanism are chosen by the program user. For user convenience, these inputs are prompted and entered interactively, after which the simulation is run with no further user action required.

A pressurized water reactor (PWR) was modeled as it is the most common type of reactor used for power plant application. As a realistic reference, the model design incorporated the applicable characteristics of the Shippingport Atomic Power Station, whose nuclear parameters were the most compatible, of available PWR core data, with the assumption of a highly enriched core made in the model design.

The model was developed by treating the reactor kinetics and other plant component thermodynamic relationships as transfer functions [1]. The point reactor kinetics equations were linearized using a Taylor expansion or small perturbation technique [2]. This same perturbation technique and lumped parameter analysis were applied to the plant's linear differential heat transfer and Xenon-135 equations [3]. The individual transfer functions were then integrated into an overall plant block diagram. The program also features a reactor protection and average reactor coolant temperature control scheme.

IBM CSMP-III language was chosen to formulate the model simulation as it has inherent routines to invert the transfer functions to the time domain.

II. MODEL DESIGN CONSIDERATIONS

Lumped parameter analysis and first order perturbation theory were used throughout the model development. All the nuclear variables which were used in the reactor kinetics and Xenon-135 decay equations were considered to be averaged values of the variables over the neutron energy spectrum. Similarly, the thermodynamic variables which were used in the plant's heat transfer equations were considered to be averaged values over the volume of the particular component.

Because of these approximations and other assumptions made in the model development, the simulation should not be considered a design or stability analysis. Instead, the simulation shows the model's large scale reactor kinetic and thermal trends during normal operating transients.

For similar reasons, although certain plant parameters of the Shippingport Atomic Power Station were incorporated into the model, the simulation cannot be considered to reflect the operating characteristics of this power plant. The Shippingport core is a seed and blanket type whereas the model core is a uniform mixture of highly enriched fuel and support materials.

A. PRESSURIZED WATER REACTOR

The pressurized water reactor (PWR) is the most widely used reactor type in central power plant applications and the only type currently used in naval propulsion.

A simplified schematic of the modeled PWR plant is shown in Figure 1. The modeled reactor contains a highly enriched core which is light

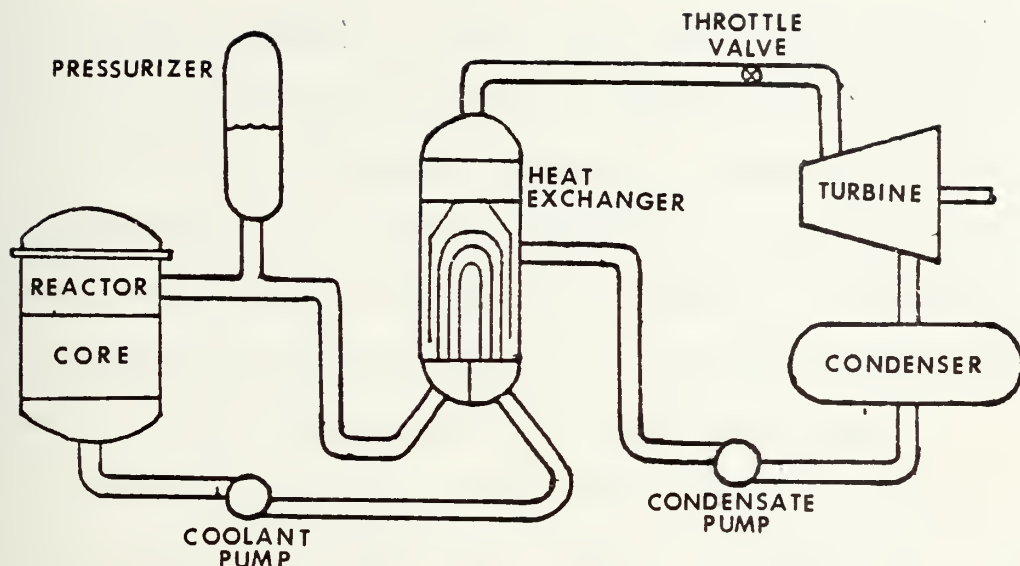


Figure 1. Schematic of the Model Plant

water moderated and cooled. The modeled plant contains two closed loop thermodynamic systems coupled by a heat exchanger.

In the primary system, heat generated from thermal fission is transferred to the coolant as it passes through the reactor, raising the coolant's temperature. The high temperature coolant leaving the reactor is circulated by a pump through tubes inside of a heat exchanger where it gives up some heat to a secondary fluid. A pressurizer maintains the primary coolant at a sufficiently high pressure so that the bulk temperature of the coolant is kept below the saturation temperature.

On the secondary side of the heat exchanger, the temperature of the entering feedwater is raised to the boiling point and saturated steam is produced. The steam is delivered to a turbine, is condensed and returned to the heat exchanger, completing the second closed loop.

B. MODEL AND PROGRAM FLEXIBILITY

The model was designed to simulate only normal operating transients from an initial steady state condition. Reactor accidents and startup were not considered in the model development. However, the model's reactor control module will simulate either a full or constant insertion of control rods if certain parameters are exceeded. This feature was not incorporated for accident analysis, but to keep parameters within the limits of the assumptions made in the model development.

The applicable Shippingport plant characteristics are fixed in the simulation program. User inputs are limited to choosing initial power level, from which other initial parameters are adjusted, and the plant perturbation mechanism, either control rod movement or turbine load change. These perturbations occur at fixed rates which limit the amount of reactivity that can be inserted during the simulation.

Originally it was envisioned that an overall program would feature not only interactive input capability but also produce real time graphical displays of the transient for a user at a time sharing computer terminal which had graphics capability. However, the computer language (IBM CSMP-III) required to solve the model's algorithms was not available on the time sharing system, and furthermore, the programs long execution time precluded any real time response.

In order to still provide as much user facility as possible with these restrictions, the interactive feature was partially retained for data input. Using a computer language available on the time sharing system (FORTRAN-IV), a program which interactively prompts the user to

enter specified controlling parameters was developed. This program incorporates logic routines which permit inputs only within requested ranges. This feature allows for data reentry if a user error is made, and ensures that only inputs compatible with the model development's assumptions are used in the simulation.

After the input is completed, the initial transient conditions are displayed at the terminal. A separate internal control program then transfers the user supplied inputs into the simulation program. The control program then transfers the simulation program to the batch processing system for execution. The interactive program notifies the user that this has been done. Hard copy plots of the transient are subsequently produced. The post-shutdown Xenon behavior module is located in the interactive program and produces real time graphical displays at the terminal.

Thus there are three distinct programs:

1. An interactive program to prompt and receive user input, and also simulate post-shutdown Xenon behavior.
2. A batch processed program which simulates the transient response to the user inputs.
3. An internal control program which interfaces these two programs.

III. MODEL DEVELOPMENT

A. POINT REACTOR KINETICS EQUATIONS

A lengthy and formal derivation of the point kinetics equations is found in Reference 4 and will not be repeated here. More generally these equations are obtained from one group neutron diffusion theory with the assumption that the neutron flux is separable in time and space, and with the inclusion of delayed neutrons [5]. They are listed below.

$$\frac{d\phi(t)}{dt} = \frac{\rho(t) - \bar{\beta}}{\Lambda} \phi(t) + \sum_i \lambda_i C_i(t) + Q(t) \quad (1)$$

$$\frac{dC_i(t)}{dt} = \frac{\bar{\beta}_i}{\Lambda} \phi(t) - \lambda_i C_i(t) \quad i = 1, 2, \dots \quad (2)$$

where $\phi(t)$ = Flux amplitude function
 $\rho(t)$ = Reactivity
 $C_i(t)$ = Effective concentration of the i th delayed neutron precursor group
 $Q(t)$ = Extraneous delayed neutron source strength
 $\bar{\beta}_i$ = Effective delayed neutron fraction from the i th group
 λ_i = Decay constant of the i th group
and $\bar{\beta} = \sum_i \bar{\beta}_i$ = Total effective delayed neutron fraction

These equations are referred to as the point reactor kinetics equations not because the reactor is considered as a single point in their application, but since spatial variations are neglected. $\bar{\beta}_i$, Λ , and λ_i are assumed to be constant.

If the neutron flux spatial variation is assumed to be time invariant, $\phi(t)$ can be considered to represent the number of neutrons in the

core. Equation (1) may then be seen as a neutron rate equation where the three terms on the right hand side represent the rate of production of prompt, delayed, and source neutrons respectively.

Equation (2) is a rate equation for the i th delayed neutron precursor group. The two terms on the right hand side represent production and decay rates respectively. The precursors are comprised of approximately thirty isotopes which historically have been divided into six groups with decay constants ranging from 0.0124 to 3.0 seconds⁻¹ for ²³⁵U. In the model, the precursors were considered to be represented by one effective group characterized by an averaged decay constant [2].

$$\lambda = \left[\frac{1}{\bar{\beta}} \sum_{i=1}^6 \frac{\bar{\beta}_i}{\lambda_i} \right]^{-1}$$

and an effective delayed neutron fraction

$$\bar{\beta} = \sum_{i=1}^6 \bar{\beta}_i$$

In equations (1) and (2), reactor power may be substituted for neutron flux if the precursor concentration is modified by $C = E_f \sum_f C_{old}$ since

$$P = E_f \sum_f \phi$$

where P = Reactor power
 E_f = Energy released per fission
 \sum_f = Macroscopic fission cross section
 ϕ = Integrated one group neutron flux

For a reactor operating at power, the source term contribution to the overall neutron population is negligible, i.e., $Q \approx 0$.

Incorporating these assumptions, equations (1) and (2) reduce to:

$$\frac{dP(t)}{dt} = \frac{\rho(t) - \bar{\beta}}{\Lambda} P(t) + \lambda C(t) \quad (3)$$

$$\frac{dC(t)}{dt} = \frac{\bar{\beta}}{\Lambda} P(t) - \lambda C(t) \quad (4)$$

1. Zero Power Reactor Transfer Function

Consider an initially critical steady state reactor at some power level P_0 at $t \leq 0$. Equations (3) and (4) each yield:

$$C_0 = \frac{\bar{\beta}}{\lambda \Lambda} P_0 \quad (5)$$

$$\text{Now let } P(t) = P_0 + \delta P(t)$$

$$C(t) = C_0 + \delta C(t) \quad (6)$$

$$\rho(t) = \rho_0 + \delta \rho(t)$$

where the zero subscript denotes the initial steady state value and the delta prefix a small perturbation imposed at $t = 0$ about this value.

Noting that $\rho_0 = 0$ for a critical reactor and substituting equations (5) and (6) into equations (3) and (4) yields for $t \geq 0$

$$\frac{d}{dt} \delta P(t) = \frac{P_0}{\Lambda} \delta \rho(t) - \frac{\bar{\beta}}{\Lambda} P(t) + \lambda \delta C(t) \quad (7)$$

$$\frac{d}{dt} \delta C(t) = \frac{\bar{\beta}}{\Lambda} \delta P(t) - \lambda \delta C(t) \quad (8)$$

where the $\frac{\delta P \delta \rho}{\Lambda}$ term in equation (7) has been neglected.

Taking the Laplace transforms of equations (7) and (8):

$$s \delta P(s) = \frac{P_0}{\Lambda} \delta \rho(s) - \frac{\bar{\beta}}{\Lambda} \delta P(s) + \lambda \delta C(s) \quad (9)$$

$$\delta C(s) = \frac{\bar{\beta}}{\Lambda} \delta P(s) - \lambda \delta C(s) \quad (10)$$

Solving equation (10) for $C(s)$, substituting this into equation (9), and rearranging, yields:

$$\frac{\delta'P(s)}{\delta\rho(s)} = \frac{P_0}{s\left(\Lambda + \frac{\beta}{s+\lambda}\right)} \quad (11)$$

This is the zero power reactor transfer function, so called because the reactor is assumed to be operating at a sufficiently low enough power that no feedback effects are realized. These effects are examined in the following sections.

Equation (11) may be represented in block diagram form as shown in Figure 2.

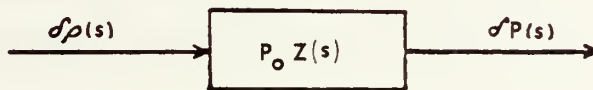


Figure 2. Block Diagram of the Zero Power Reactor Transfer Function

$$\text{where } Z(s) = \frac{1}{s\left(\Lambda + \frac{\beta}{s+\lambda}\right)}$$

2. Limitations of the Point Reactor Kinetics Equations

The point reactor kinetics equations derivation is based on the assumption that the spatial dependence of the neutron flux is negligible. This assumption limits the validity of these equations to transients where this remains a reasonable approximation such as those which result in only small changes in reactivity. The small perturbation technique used in the development of the zero power reactor

transfer function also requires only small changes in reactivity if the first order approximation is to hold. Specifically, ρ must be less than $0.5\bar{\beta}$. Physically, when ρ is greater than $\bar{\beta}$ the reactor is critical on prompt neutrons alone. The simulated transients imposed on the model were limited to ensure that excessive amounts of reactivity were not introduced.

B. REACTIVITY FEEDBACK MECHANISMS

In the zero power reactor point reactor, the power level is assumed to be so low that it does not affect the reactivity, thus there are no feedback effects. However, for a reactor operating at a useful power level, feedback effects do exist and the reactivity becomes an implicit function of the reactor power level (or neutron flux). This dependence arises since reactivity depends on macroscopic cross sections which involve the atomic number densities of material in the core:

$$\Sigma = N\sigma$$

where Σ = Macroscopic cross section (cm^{-1})
 N = Atomic number density (atoms/cm^3)
 σ = Microscopic cross section (cm^2)

The atomic number density can depend upon the reactor power level since the concentrations of certain nuclei are constantly changing due to neutron interactions. Material densities also depend upon temperature which is a function of reactor power level and hence the flux.¹

¹Duderstadt, J. J. and Hamilton, L. J., Nuclear Reactor Analysis, 1st ed., Wiley, 1976.

The two feedback mechanisms considered in the model were Xenon-135 and moderator temperature.

Reactivity can also be changed directly by an external source such as control rods containing a neutron absorbing material. Thus the overall reactivity can be written as:

$$\delta\rho = \delta\rho_E + \delta\rho_X + \delta\rho_T \quad (12)$$

where $\delta\rho_X = F_X P$
 $\delta\rho_T = F_T P$
 $\delta\rho$ = Overall core reactivity
 $\delta\rho_E$ = Externally added reactivity
 $\delta\rho_X$ = Xenon-135 feedback reactivity
 $\delta\rho_T$ = Moderator temperature feedback reactivity
 F_X = Xenon transfer function
 F_T = Moderator temperature transfer function

Equation (12) with the previously developed zero power reactor transfer function, given by equation (11), are incorporated in the block diagram below:

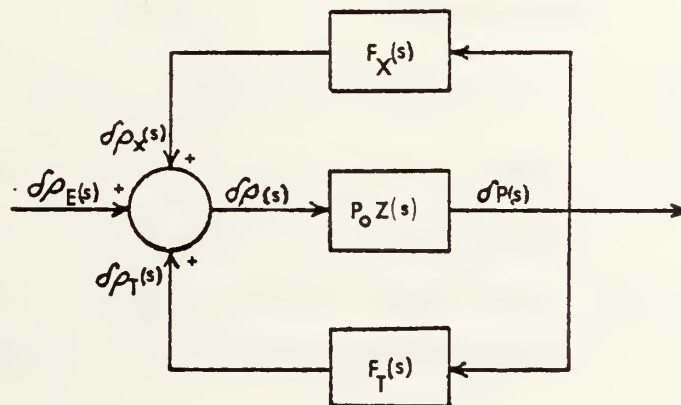


Figure 3. Block Diagram of the Reactor Transfer Function with Xenon-135 and Moderator Temperature Feedback Loops, and External Reactivity

The next step in the model development was to derive the Xenon-135 and moderator temperature transfer functions.

1. Xenon-135 Transfer Function

Xenon-135 is the most significant fission product poison because of its enormous thermal neutron absorption cross section and relatively large fission yield. This isotope is not only produced directly from fission but also from the decay of other fission products as shown in Figure 4.

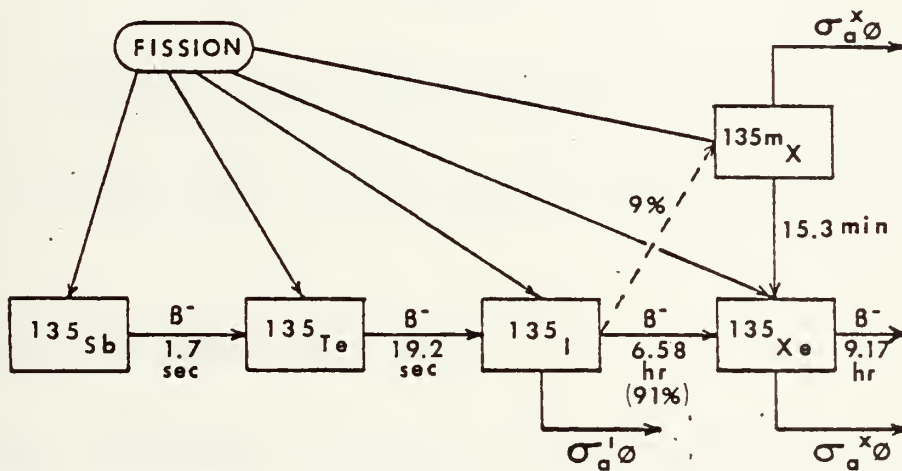


Figure 4. Xe-135 Decay Scheme

Since the β^- decay of Iodine-135 (^{135}I) and Xenon-135 (^{135}Xe), with the largest half-lives, are the controlling steps in this decay scheme, it was simplified by making the following assumptions:

- 1) All ^{135}I is produced directly from fission (the production of Antimony-135 (^{135}Sb) and subsequent decay to ^{135}I is considered instantaneous).
- 2) The short lived metastable $^{135\text{m}}\text{Xe}$ is ignored.
- 3) The removal of ^{135}I by neutron absorption is negligible for the neutron flux levels used in the model (10^{14} neutrons/cm²sec).

4) All the ^{135}I decays to ^{135}Xe .

With these assumptions the effective decay scheme is shown in Figure

5.

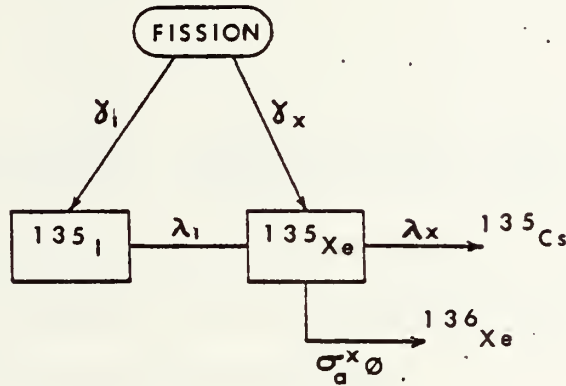


Figure 5. Simplified Xe-135 Decay Scheme

Using this decay scheme, the resulting rate equations for ^{135}I and ^{135}Xe are:

$$\frac{dI(t)}{dt} = \gamma_I \Sigma_f \phi(t) - \lambda_I I(t) \quad (13)$$

$$\frac{dX(t)}{dt} = \gamma_X \Sigma_f \phi(t) + \lambda_I I(t) - \lambda_X X(t) - \sigma_a^X \phi(t) X(t) \quad (14)$$

where $I = ^{135}\text{I}$ number density (atoms/cm³)

$X = ^{135}\text{Xe}$ number density (atoms/cm³)

Σ_f = Macroscopic fission cross section (cm⁻¹)

ϕ = Integrated one group neutron flux (neutrons/cm²sec)

γ_I = Effective ^{135}I fission yield

γ_X = Effective ^{135}Xe fission yield

λ_I = ^{135}I decay constant (sec⁻¹)

λ_X = ^{135}Xe decay constant (sec⁻¹)

σ_a^X = ^{135}Xe microscopic thermal neutron absorption cross section (cm²)

Again, using the first order perturbation technique, let

$$\begin{aligned} I(t) &= I_0 + \delta I(t) \\ X(t) &= X_0 + \delta X(t) \\ \phi(t) &= \phi_0 + \delta \phi(t) \end{aligned} \quad (15)$$

where the zero subscript denotes an initial steady state value and the delta prefix denotes a small perturbation about this value.

Upon substituting equations (15) into equations (13) and (14), and taking the Laplace transform, the relationship between the perturbation in ^{135}Xe and ϕ is derived (See Appendix A):

$$\frac{\delta X(s)}{\delta \phi(s)} = \frac{(\lambda_X \Sigma_f - \sigma_a^X X_0)s + \lambda_I (\lambda_I \Sigma_f + \lambda_X \Sigma_f - \sigma_a^X X_0)}{s^2 + (\sigma_a^X \phi_0 + \lambda_X + \lambda_I)s + \lambda_I (\sigma_a^X \phi_0 + \lambda_X)} = G_X(s) \quad (16)$$

The change in reactivity caused by a small perturbation in ^{135}Xe concentration is also derived in Appendix A:

$$\frac{\delta \rho_X(s)}{\delta X(s)} = - \frac{1}{X_0 + \frac{\sigma_a^F}{\sigma_a^X} U} = \alpha_X \quad (17)$$

where X_0 = Equilibrium ^{135}Xe number density before the perturbation (atoms/cm³)

U = Uranium-235 (^{235}U) number density (atoms/cm³)

σ_a^F = Microscopic thermal neutron absorption cross section of ^{235}U (cm²)

A perturbation in neutron flux ($\delta \phi$) is directly proportional to a perturbation in reactor power (δP) as shown by:

where E_f = Energy released per fission

Σ_f = Macroscopic fission cross section of ^{235}U (cm^{-1})

or

$$\frac{\delta\phi(s)}{\delta P(s)} = \frac{1}{E_f \Sigma_f} = K \quad (18)$$

The product of equations (17), (16), and (18) yields:

$$\frac{\delta\rho_x(s)}{\delta X(s)} \frac{\delta X(s)}{\delta\phi(s)} \frac{\delta\phi(s)}{\delta P(s)} = \frac{\delta\rho_x(s)}{\delta P(s)} = \alpha_x G_x(s) K = F_x(s) \quad (19)$$

Equation (19) is the transfer function relating reactor power and ^{135}Xe feedback reactivity. This equation is shown in block diagram form in Figure 6 where it has been incorporated with the previously derived zero power transfer function given by equation (11).

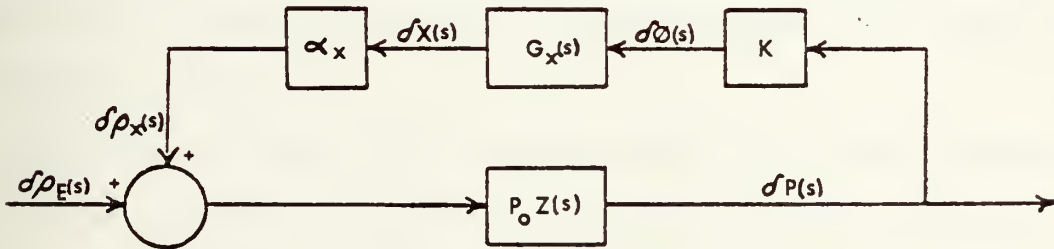


Figure 6. Block Diagram of the Reactor Transfer Function with Xenon-135 Feedback and External Reactivity

2. Moderator Temperature Feedback

Reactivity feedback from changes in moderator temperature occurs as a result of changes with moderator density. The density is also a

function of pressure, however, the pressure coefficient of reactivity is typically two orders of magnitude smaller than the temperature coefficient, and was therefore not considered in the model development. The moderator density affects the moderator number density and hence the macroscopic scattering cross section of the moderator.

The primary mechanism for the thermalization of the prompt and delayed neutrons is by elastic scattering interactions with the moderator nuclei. Hence, variations in the macroscopic scattering cross section will affect the rate at which neutrons become thermalized. Changes in the thermalization rate affect the fission rate, or reactor power level. Power level changes affect the moderator temperature, thus a feedback loop is created.

Because the moderator is also the coolant in the model, its temperature is not only a function of reactor power but also a function of the heat transfer process occurring in the heat exchanger, thus complicating the feedback loop.

Therefore, in order to develop this feedback mechanism analytically, it is first necessary to model the plant's heat transfer processes. This thermal analysis is done in the following section.

Another temperature feedback mechanism is the broadening of the Uranium-238 resonance absorption cross section for neutrons with increasing temperature. Because a highly enriched core was assumed in the model, with little Uranium-238, this effect was not considered.

C. THERMAL ANALYSIS

1. Reactor Heat Transfer Function

A lumped parameter model was assumed. This simplification provided a set of ordinary differential equations which were sufficiently accurate for the simulated normal operating transients. In the lumped parameter model, heat transfer in the reactor was assumed to occur at a single point. Thus, spatial variations were neglected. The core was considered to be a homogenized mixture of the uranium alloy fuel, the fuel cladding, and other structural materials, with a constant thermal capacity.

The equation for the heat flow from the core was obtained from a basic heat balance. The heat generated from fission equals the heat required to change the temperature of the core materials plus the heat transferred to the coolant. On a per unit time basis, the heat balance is:

$$P(t) = C_F \frac{dT_F(t)}{dt} + h_{FM} \left[T_F(t) - T_{AV}(t) \right] \quad (20)$$

where $P(t)$ = Total Power generated in the core (Btu/sec)

$T_F(t)$ = Average temperature of the core materials ($^{\circ}\text{F}$)

$T_{AV}(t)$ = Average reactor coolant temperature ($^{\circ}\text{F}$)

C_F = Total thermal capacity of the core materials (Btu/ $^{\circ}\text{F}$)

h_{FM} = Total heat transfer coefficient (Btu/ $^{\circ}\text{F sec}$)

A similar heat transfer balance must also hold for the heat being transferred to the coolant and transported out of the core. This heat is the last term of equation (20) and is transferred to the coolant as:

$$h_{FM} [T_F(t) - T_{AV}(t)] = C_M \frac{dT_{AV}(t)}{dt} + \dot{m}_m C [T_{H_o}(t) - T_{C_i}(t)] \quad (21)$$

where C_M = Total thermal capacity of coolant in the core (Btu/°F)
 \dot{m}_m = Coolant mass flow rate (lbm/sec)
 T_{H_o} = Average reactor coolant outlet temperature (°F)
 T_{C_i} = Average reactor coolant inlet temperature (°F)
 C = Specific heat of reactor coolant (Btu/lbm°F)

For simplification, the average reactor coolant temperature is assumed to be given by:

$$T_{AV}(t) = [T_{C_i}(t) + T_{H_o}(t)] / 2 \quad (22)$$

Rearranging equation (20) yields:

$$T_F(t) + \tau_1 \frac{dT_F(t)}{dt} = T_{AV}(t) + \frac{\tau_1}{C_F} P(t) \quad (23)$$

where $\tau_1 = \frac{C_F}{h_{FM}}$

Substituting equation (22) into equation (21) to eliminate T_{H_o} yields:

$$T_{AV}(t) \left[1 + 2 \frac{\tau_2}{\tau_o} \right] + \tau_2 \frac{dT_{AV}(t)}{dt} = T_F(t) + 2 \frac{\tau_2}{\tau_o} T_{C_i}(t) \quad (24)$$

where $\tau_o = \frac{C_M}{\dot{m}_m C}$, and
 $\tau_2 = \frac{C_M}{h_{FM}}$

Now let

$$\begin{aligned}
 P(t) &= P_0 + \delta P(t) \\
 T_F(t) &= T_{F_0} + \delta T_F(t) \\
 T_{AV}(t) &= T_{AV_0} + \delta T_{AV}(t) \\
 T_{Ho}(t) &= T_{Ho_0} + \delta T_{Ho}(t) \\
 T_{Ci}(t) &= T_{Ci_0} + \delta T_{Ci}(t)
 \end{aligned} \tag{25}$$

where the zero subscript denotes a steady state value and the prefix a small perturbation about this value.

Substituting equations (25) into equations (22), (23), and (24), eliminating the average temperatures $T_F(t)$ and $T_{AV}(t)$, and taking Laplace transforms, the following relationship between reactor coolant inlet and outlet temperature perturbations is derived (see Appendix B):

$$\delta T_{Ho}(s) = \frac{- \left\{ \frac{\tau_1 \tau_2}{2} s^2 + \left[\frac{\tau_0}{\tau_2} \left(\frac{\tau_1}{\tau_2} + 1 \right) - \tau_1 \right] s - 1 \right\} \delta T_{Ci}(s) + \gamma \delta P(s)}{\left\{ \frac{\tau_1 \tau_2}{2} s^2 + \left[\frac{\tau_0}{\tau_2} \left(\frac{\tau_1}{\tau_2} + 1 \right) + \tau_1 \right] s + 1 \right\}} \tag{26}$$

where

$$\gamma = \frac{\tau_1 \tau_0}{C_F \tau_2} = \frac{1}{\dot{m}_H C}$$

This equation is the reactor heat transfer function for the reactor coolant outlet temperature as a function of the reactor coolant inlet temperature and reactor power.

A block diagram representation of this transfer function is shown in Figure 7.

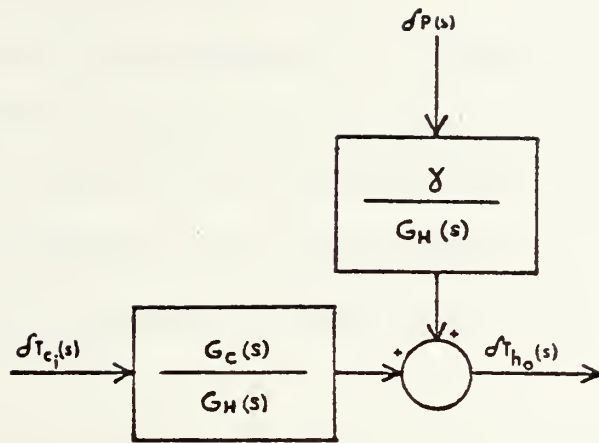


Figure 7. Block Diagram of the Reactor Heat Transfer Function

In Figure 7

$$G_C(S) = \left\{ \frac{\tau_1 \tau_2}{2} s^2 + \left[\frac{\tau_0}{\tau_2} \left(\frac{\tau_1}{\tau_2} + 1 \right) - \tau_1 \right] s + 1 \right\}$$

$$G_H(S) = \frac{1}{\left\{ \frac{\tau_1 \tau_2}{2} s^2 + \left[\frac{\tau_0}{\tau_2} \left(\frac{\tau_1}{\tau_2} + 1 \right) + \tau_1 \right] s + 1 \right\}}$$

2. Heat Exchanger Transfer Function

As in the derivation of the reactor heat transfer function, a lumped parameter model was assumed for the heat exchanger. Two points of energy storage were assumed, the primary coolant water and the water on the secondary side of the heat exchanger. The thermal capacity of the heat exchanger metal was included with that for the secondary water, since the thermal resistance on the primary side is predominant. A further assumption was made that the time spent by the primary coolant

while it passed through the heat exchanger was negligible in comparison with the time spent in the primary piping.

The state of the steam produced on the secondary side of the heat exchanger was assumed to always be dry and saturated and that the secondary water is always at the saturation temperature for the existing pressure. These assumptions were justified because moisture separators and recirculation can provide high quality steam and preheating of the feedwater.

With these assumptions, the following equations were obtained from a heat balance per unit time:

$$\dot{m}_M C [T_{Hi}(t) - T_{Co}(t)] = C_M \frac{dT_{AVB}(t)}{dt} + h_{TM} [T_{AVB}(t) - T_S(t)] \quad (27)$$

$$h_{TM} [T_{AVB}(t) - T_S(t)] = C_S \frac{dT_S(t)}{dt} + P_L(t) \quad (28)$$

- where
- $T_{Hi}(t)$ = Heat exchanger coolant inlet temperature ($^{\circ}\text{F}$)
 - $T_{Co}(t)$ = Heat exchanger coolant outlet temperature ($^{\circ}\text{F}$)
 - $T_{AVB}(t)$ = Heat exchanger average coolant temperature ($^{\circ}\text{F}$)
 - $T_S(t)$ = Saturated steam temperature ($^{\circ}\text{F}$)
 - h_{TM} = Total heat transfer coefficient ($\text{BTU}/^{\circ}\text{F}.\text{sec}$)
 - C_M = Total thermal capacity of coolant in the heat exchanger ($\text{BTU}/^{\circ}\text{F}$)
 - C_S = Total thermal capacity of heat exchanger metal and secondary water and steam ($\text{Btu}/^{\circ}\text{F}$)
 - $P_L(t)$ = Power delivered by the heat exchanger (BTU/sec)
 - \dot{m}_M = Coolant mass flow rate (lbm/sec)

The total heat transfer coefficient, a function of the primary coolant flow rate (a constant in the model) and the heat transfer characteristics of the heat exchanger, was assumed to be constant. Also, the time delay in transferring heat across the heat exchanger tubes was neglected. This changed the shape of the initial thermal transient but had little effect on the basic dynamics of the secondary loop.

Again, as a simplification, the average temperature of the coolant in the heat exchanger was assumed to be given by

$$T_{AV_B} = \frac{1}{2} (T_{H_i} + T_{C_o}) \quad (29)$$

The power delivered by the steam generator is proportional to the product of the steam flow rate and the difference in enthalpy between the steam and feedwater

$$h_{TM} [T_{AV_B}(t) - T_S(t)] = \dot{m}_S (H_S - H_{FW}) + C_S \frac{dT_S(t)}{dt}$$

where \dot{m}_S = Steam flow rate (lbm/sec)
 H_S = Saturated steam enthalpy (Btu/lbm)
 H_F = Feedwater enthalpy (Btu/lbm)

This equation assumes the steam and feedwater flow rates are always equal and thus neglects any instabilities in the secondary steam.

As discussed in Reference 4, the enthalpy of saturated steam is nearly constant over a wide range of pressure, varying from 1198 Btu/lbm to 1204 Btu/lbm over a pressure range from 200 to 800 psig. The feedwater enthalpy, which depends on condenser pressure, is usually between 50 and 100 Btu/lbm. Thus the enthalpy difference may be regarded as a

constant, and the power delivered by the heat exchanger is directly proportional to the steam flow rate.

The impedance to steam flow caused by the turbine is nearly independent of turbine speed. If constant backpressure is assumed, the steam flow rate is directly proportional to the throttle opening at a given pressure. Thus,

$$\dot{m}_s = P_s A$$

where P_s = Saturated steam pressure (lbf/in²)

A = Proportionality factor which is a function
of the throttle setting $\frac{1 \text{bm in}^2}{1 \text{bf sec}}$

The numerous assumptions made in the heat exchanger model development limit the accuracy of the resulting equations. However, the errors involved in these assumptions are usually less than the amount of uncertainty in the engineering values of the coefficients used in the equations.

Recalling the previous assumption that

$$\frac{dT_{AV_B}(t)}{dt} = 0, \text{ and}$$

Substituting equation (29) into equation (27), yields:

$$\dot{m}_M C [T_{H_i}(t) - T_{C_o}(t)] = h_{TM} \left\{ \frac{1}{2} [T_{H_i}(t) + T_{C_o}(t)] - T_S(t) \right\} \quad (30)$$

Solving equation (30) for T_{C_o} , gives:

$$T_{C_o}(t) = \frac{2T_S(t) - T_{H_i}(1 - K_1)}{(1 + K_1)} \quad (31)$$

where $K_1 = \frac{2 \dot{m}_M C}{h_{TM}}$

Substituting equation (29) into equation (28) gives:

$$h_{TM} \left\{ \frac{1}{2} [T_{Hi}(t) + T_{Co}(t)] - T_S(t) \right\} = C_S \frac{dT_S(t)}{dt} + P_L(t)$$

Rearranging this expression,

$$\frac{C_S}{h_{TM}} \frac{dT_S(t)}{dt} + T_S(t) = \frac{1}{2} [T_{Hi}(t) + T_{Co}(t)] + \frac{1}{h_{TM}} P_L(t) \quad (32)$$

Let

$$\begin{aligned} T_{Hi}(t) &= T_{Hi0} + \delta T_{Hi}(t) \\ T_{Co}(t) &= T_{Co0} + \delta T_S(t) \\ T_S(t) &= T_{X0} + \delta T_S(t) \\ P_L(t) &= P_{L0} + \delta P_L(t) \end{aligned} \quad (33)$$

where the lowest zero subscript denotes a steady state value and the delta prefix a small perturbation about this value.

By substituting equations (33) into equations (31) and (32), and taking the Laplace transform, the following set of equations is derived (see Appendix C):

$$\delta T_S(s) = \frac{\frac{1}{2} [\delta T_{Co}(s) + \delta T_{Hi}(s)] - \frac{1}{h_{TM}} \delta P_L(s)}{\tau'_S + 1} \quad (34)$$

$$\delta T_{Co}(s) = \frac{2 \delta T_S(s) - \delta T_{Hi}(s) [1 - K_1]}{1 + K_1} \quad (35)$$

where $\tau'_S = \frac{C_S}{h_{TM}}$

These equations are represented in a block diagram in Figure 8.

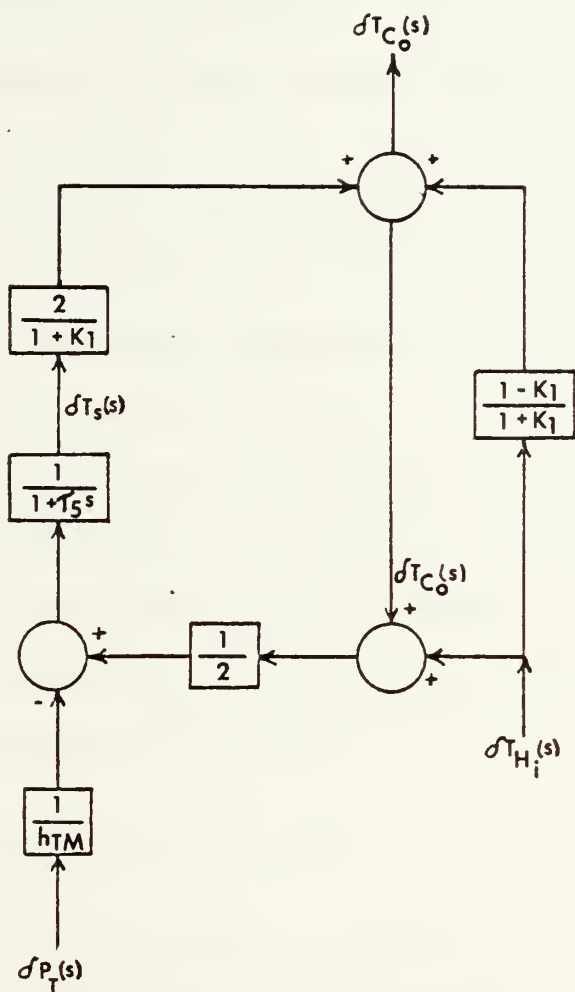


Figure 8. Block Diagram of the Heat Exchanger Transfer Function

With δT_{H_i} and δP_L as inputs, equations (34) and (35) form an algebraic loop in δT_{C_o} and δT_s . These two equations were solved with the use of an inherent functional routine available in the CSMP-III language which was used to formulate the model.

3. Primary Piping Transfer Functions

While circulating through the primary loop, the coolant undergoes mixing and transport delay effects. For example, a transient in the coolant temperature at the heat exchanger outlet does not appear at the reactor inlet until sometime later. Where there are volume or flow direction changes, as in the reactor coolant inlet plenum, mixing occurs causing a smearing of a temperature transient. Both of these effects were approximated in the model by combinations of two types of time delays: a pure transport delay and a simple mixing delay.

Assuming no mixing in the primary piping and no heat loss with perfectly insulated pipes, pure transport delays are encountered in the piping runs between the reactor and heat exchanger. With these assumptions, the temperatures involved in the transfer of heat between the reactor and heat exchanger can be given as

$$T_{H_i}(t) = T_{H_o}(t - \tau_3)$$

where the inlet coolant temperature to the heat exchanger T_{H_i} has the same form as the outlet temperature of the reactor T_{H_o} after a fixed transport delay τ_3 . This transport delay can be approximated by the following differential equation which is derived in Appendix D.

$$T_{H_i} + \tau_3 \frac{dT_{H_i}}{dt} = T_{H_o} \quad (36)$$

Similarly, the transport delay from the heat exchanger outlet to the reactor inlet plenum is:

$$T_{C_{ip}}(t) + \tau_4 \frac{dT_{C_{ip}}(t)}{dt} = T_{C_o}(t) \quad (37)$$

where $T_{C_{ip}}(t)$ = Reactor inlet plenum coolant temperature ($^{\circ}\text{F}$)
 $T_{C_o}(t)$ = Heat exchanger outlet coolant temperature ($^{\circ}\text{F}$)

This can be approximated by

$$T_{C_{ip}}(t) + \tau_4 \frac{dT_{C_{ip}}(t)}{dt} = T_{C_o}(t) \quad (37)$$

As in the previous development, let

$$\begin{aligned} T_{H_o}(t) &= T_{H_{o_o}} + \delta T_{H_o}(t) \\ T_{H_{ip}}(t) &= T_{H_{ipo}} + \delta T_{H_{ip}}(t) \\ T_{C_o}(t) &= T_{C_{o_o}} + \delta T_{C_o}(t) \\ T_{C_{ip}}(t) &= T_{C_{ipo}} + \delta T_{C_{ip}}(t) \end{aligned} \quad (38)$$

where again the lowest zero subscript denotes a steady state value and the delta prefix a small perturbation about this value.

After substituting equations (38) into equations (36) and (37), taking the Laplace transform, the following transfer functions are derived (see in Appendix D).

$$\frac{\delta T_{H_{ip}}(s)}{\delta T_{H_o}(s)} = \frac{1}{1 + \tau_3 s} \quad (39)$$

$$\frac{\delta T_{C_{ip}}(s)}{\delta T_{C_o}(s)} = \frac{1}{1 + \tau_4 s} \quad (40)$$

Combined mixing and transport effects are encountered at the reactor and heat exchanger inlet and outlet coolant plenums. For simplification, only the mixing effects at the inlet plenums were considered in the model. Assuming perfect mixing and after performing a heat balance on the plenum concerned, the combined mixing and transport effects are expressed by differential equations developed in Appendix D of the form

$$\tau \frac{dT_o(t)}{dt} + T_o(t) = T_i(t)$$

where

$T_i(t)$ = Plenum inlet temperature (°F)

$T_o(t)$ = Plenum outlet temperature (°F)

τ = Mixing delay

As before, by using a small perturbation technique, and taking the Laplace transform, the following transfer function is derived (see Appendix D):

$$\frac{\delta T_{Ho}(s)}{\delta T_{Hi}(s)} = \frac{1}{1 + \tau s} \quad (41)$$

Block diagram representations of the coolant piping transport and mixing transfer functions is shown in Figure 9.

D. OVERALL PLANT BLOCK DIAGRAM

The transfer functions developed previously for the zero power point reactor, ^{135}Xe feedback, reactor and heat exchanger heat transfer, and the primary piping were interconnected resulting in the overall model block diagram shown in Figure 10. The moderator temperature feedback

1. General Parameters

- a. Reactor thermal power 231 MW
- b. Reactor coolant system pressure 2000 psia
- c. Reactor coolant average temperature 523° F
- d. Steam pressure at full load 600 psia

2. Reactor Coolant System

- a. Reactor coolant flow rate 6280 lbm/sec
- b. Reactor coolant outlet temperature 538° F
- c. Reactor coolant inlet temperature 508° F
- d. Coolant volume in core 103 ft³

3. Reactor Core

- a. Configuration Right cylinder
- b. Size 6.8 ft. dia X 6 ft. high
- c. Fuel load U²³⁵ (seed) 75 kg
- d. Composition (seed)
 - 1) Water 43.5 v/o
 - 2) Fuel alloy 30.0 v/o
 - 3) Zircalloy 34.1 v/o
- e. Control Rods
 - 1) Total rod worth 0.256 k
 - 2) Scram time 0.35 sec time delay
1.0 sec rod drop

4. Nuclear Data

- a. Thermal neutron flux 2×10^{14} n/cm² sec
- b. Prompt neutron lifetime 5.6×10^{-5} sec

- c. Effective delayed neutron fraction 0.0077
- d. Temperature coefficient of reactivity $-3.1 \times 10^{-4} \text{ } \rho_k / ^\circ\text{F}$

5. Reactor Protection Setpoints

a. Scram

- 1) High reactor power 138%
- 2) High reactor coolant outlet temperature 550°F

b. Cutback

- 1) High reactor power 114%
- 2) High Startup rate 1.74 Decades per min.

IV. RESULTS

A. INTERACTIVE PROGRAM

Figures 10-16 are examples of the interactive program output. As shown in Figure 10, the user is first given a description of the purpose of the overall program and general instructions for input entry. As shown in Figure 11, the user is then prompted to enter an initial steady state power level, after which the corresponding major plant parameters are displayed. The user is then prompted to choose the type of simulation, either plant transient at power, or post-shutdown Xenon-135 behavior. The user must be at the Tektronix 4012 console in the computer center if the latter is chosen.

If the post-shutdown Xenon-135 behavior is chosen to be examined, graphs such as those shown in Figure 13 are generated and displayed on the Tektronix's screen. In addition to showing the time response of the Xenon, the time of its peak and associated maximum reactivity are displayed.

If the plant transient simulation is chosen, the user is then prompted to choose the mechanism for initiating the transient, either control rod movement or turbine load change. In Figure 14, the user has chosen the former. The program then prompts the user to enter the direction and time length of the control rod movement. In Figure 15, the turbine load change was chosen and the program request the final turbine load. In either case, a summary of the transient inputs is displayed with a notice that the interactive portion is complete.

EXECUTION BEGINS....

THIS IS AN INTERACTIVE PROGRAM WHICH SIMULATES THE THERMAL RESPONSE OF A 231 MWt (68 MWe), HIGHLY ENRICHED PRESSURIZED WATER REACTOR (PWR), POWER PLANT TO CHANGES IN REACTIVITY INITIATED BY CHANGES IN BANK CONTROL ROD POSITION OR TURBINE LOAD. THE REACTIVITY FEEDBACK EFFECTS RESULTING FROM CHANGES IN MODERATOR TEMPERATURE AND XENON-135 CONCENTRATION ARE INCORPORATED INTO THE MODEL.

PLEASE ANSWER THE FOLLOWING STATEMENTS BY ENTERING THE NUMBER (INCLUDING THE DECIMAL POINT, BUT NOT THE PERCENT SYMBOL) WHICH CORRESPONDS TO THE DESIRED CHOICE OR VALUE WITHIN THE GIVEN RANGE.

DO YOU WANT A BRIEF DESCRIPTION OF THE PLANT?

1. YES
2. NO

>2.

Figure 11. Interactive Program, Introduction and Instructions

ENTER AN INITIAL STEADY STATE POWER LEVEL BETWEEN 10. AND 100. PERCENT.
NOTE: A MINIMUM OF 10% POWER IS REQUIRED TO OPERATE THE REACTOR COOLANT
PUMPS AND OTHER PLANT AUXILIARIES
>50.

INITIAL STEADY STATE PARAMETERS:

REACTOR POWER= 50.00% 115.50 MWE
THERMAL NEUTRON FLUX= 0.10E 15 NEUTRONS/CM2 SEC
XENON-135 CONCENTRATION= 0.76E 12 ATOMS/CM3
FUEL TEMPERATURE=711.50 F
AVERAGE REACTOR COOLANT MODERATOR TEMPERATURE=523.00 F
REACTOR COOLANT OUTLET TEMPERATURE=530.50 F
REACTOR COOLANT INLET TEMPERATURE=515.50 F
SATURATED STEAM TEMPERATURE=504.54 F
SATURATED STEAM PRESSURE=704.07 PSIA
TURBINE LOAD= 50.00% 34.00 MWE

ENTER THE DESIRED SIMULATION:

1. MODERATOR TEMPERATURE AND XENON FEEDBACK AT POWER
 2. POST SHUTDOWN XENON BEHAVIOR
- >1.

Figure 12. Interactive Program, Initial Power Level and Simulation Choice Inputs

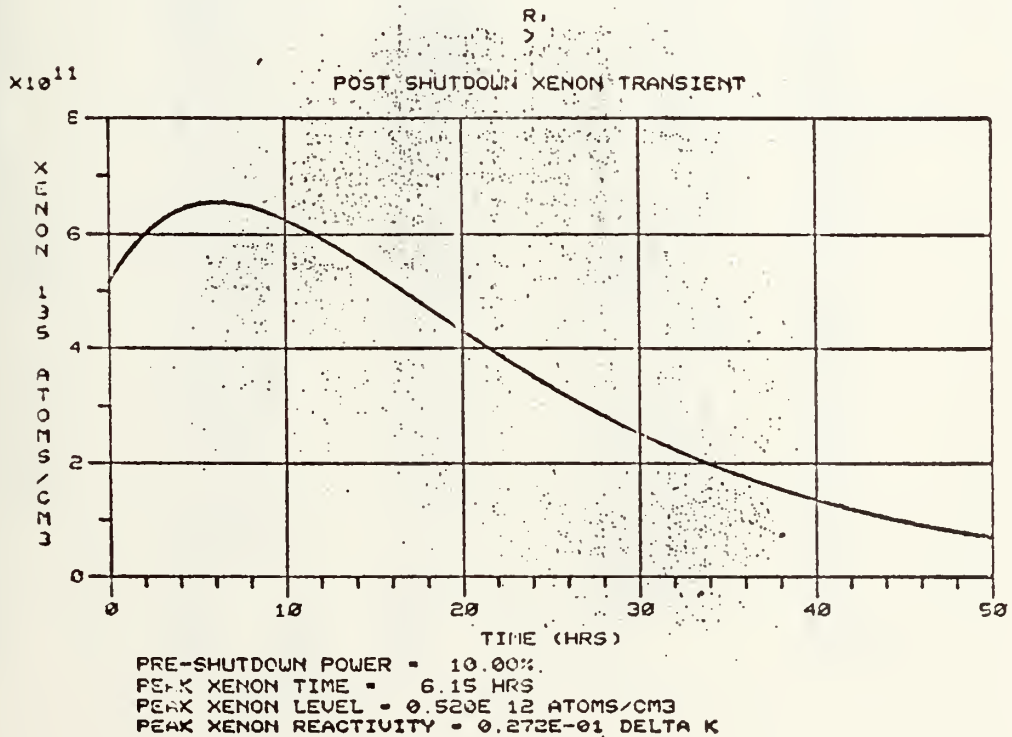
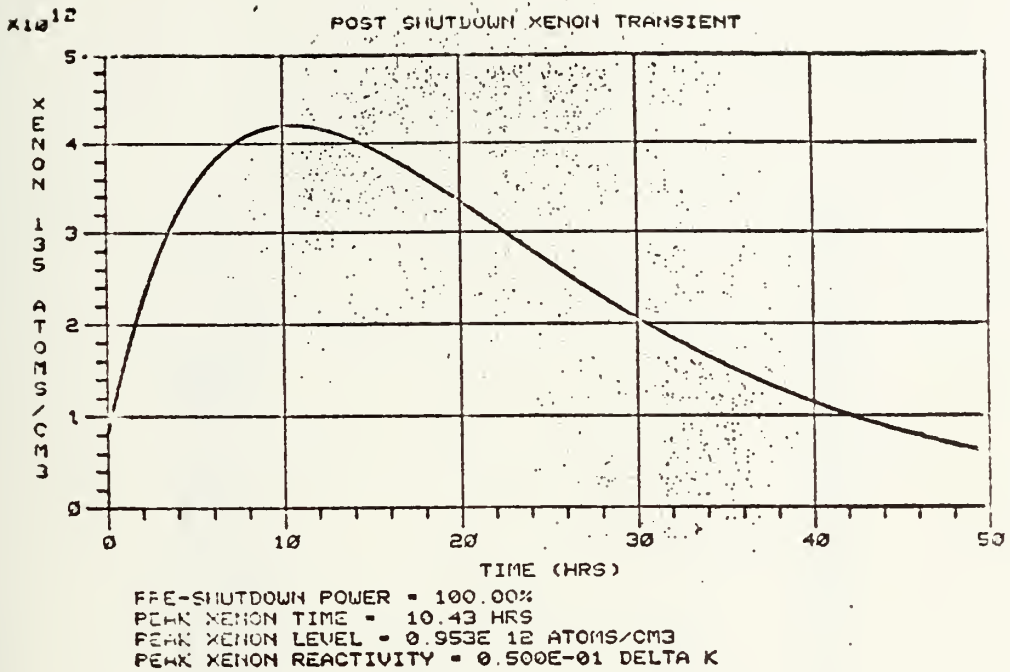


Figure 13. Post-shutdown Xenon-135 Behavior

ENTER THE INITIAL REACTIVITY CHANGE MECHANISM:

1. BANK CONTROL ROD CHANGE
2. TURBINE LOAD CHANGE

>1.

ENTER THE TIME LENGTH (SEC) OF BANK CONTROL ROD MOVEMENT (TSHIM)
NOTE: BANK CONTROL ROD REACTIVITY INSERTION RATE IS 1.25E-4 DELTA K/SEC,
DO NOT EXCEED A TOTAL REACTIVITY INSERTION OF 35.0E-4 DELTA K
(I.E. A 28 SEC SHIM)

>10.

ENTER THE DIRECTION OF BANK CONTROL ROD SHIM:

1. IN (NEGATIVE REACTIVITY INSERTION)
2. OUT (POSITIVE REACTIVITY INSERTION)

>2.

INITIAL REACTIVITY CHANGE PARAMETERS:

REACTOR POWER = 50.00 %

SHIM TIME = 10.00 SEC

SHIM DIRECTION = 1.00 (1="OUT", -1="IN")

FINAL TURBINE LOAD = 50.00%

INPUTS HAVE BEEN ENTERED INTO THE SIMULATION PROGRAM WHICH HAS BEEN SENT
TO THE BATCH SYSTEM FOR EXECUTION

R;
>

Figure 14. Interactive Program, Control Rod Movement Inputs

ENTER THE INITIAL REACTIVITY CHANGE MECHANISM:

1. BANK CONTROL ROD CHANGE
2. TURBINE LOAD CHANGE

>2.

ENTER FINAL TURBINE LOAD BETWEEN 10. AND 100. PERCENT

NOTE: LOAD VARIATION WILL OCCUR AT A FIXED RATE OF 0.5%/SEC
>75.

INITIAL REACTIVITY CHANGE PARAMETERS:

REACTOR POWER = 50.00 %

SHIM TIME = 0.0 SEC

SHIM DIRECTION = -0.25 (1="OUT", -1="IN")

FINAL TURBINE LOAD = 75.00%

INPUTS HAVE BEEN ENTERED INTO THE SIMULATION PROGRAM WHICH HAS BEEN SENT
TO THE BATCH SYSTEM FOR EXECUTION

R;

>

Figure 15. Interactive Program, Turbine Load Change Inputs

Figure 16 is an example of the interactive program's logic routines which will only permit inputs within the requested ranges to be accepted for the simulation.

B. SIMULATION PROGRAM

The program generates Versatec plots showing the plant's power, reactivity, and temperature transient responses for the user's inputs. Figures 17-20 are composites of these plots.

In Figure 17, the transient was initiated by a simulated 20 to 40 percent ramp change in turbine load at 1/2 percent per second. In Figure 18, the transient was initiated by a simulated 60 to 40 percent change in turbine load in the same manner. Both of these figures show the characteristic power demand following and inherent stability response typical of a PWR plant with a constant average temperature program and negative temperature coefficient.

The inherent stability feature is further displayed in Figures 19 and 20. In Figure 19, the transient is initiated by a simulated 10 second inward movement of the control rods at a reactivity insertion rate of $1.25 \times 10^{-4} \text{ } \rho_k$ per second from an initial 50 percent power level. In Figure 20, the transient response to an outward control rod movement with the same parameters is shown. As seen in both cases, the reactor power stabilizes about the initial turbine load and the average reactor coolant temperature stabilizes at a new level to compensate, via the negative temperature coefficient, for the reactivity inserted by the control rods.

ENTER AN INITIAL STEADY STATE POWER LEVEL BETWEEN 10. AND 100. PERCENT.
NOTE: A MINIMUM OF 10% POWER IS REQUIRED TO OPERATE THE REACTOR COOLANT
PUMPS AND OTHER PLANT AUXILIARIES
>150.

INCORRECT DATA ENTERED

ENTER AN INITIAL STEADY STATE POWER LEVEL BETWEEN 10. AND 100. PERCENT.
NOTE: A MINIMUM OF 10% POWER IS REQUIRED TO OPERATE THE REACTOR COOLANT
PUMPS AND OTHER PLANT AUXILIARIES
>5.

INCORRECT DATA ENTERED

ENTER AN INITIAL STEADY STATE POWER LEVEL BETWEEN 10. AND 100. PERCENT.
NOTE: A MINIMUM OF 10% POWER IS REQUIRED TO OPERATE THE REACTOR COOLANT
PUMPS AND OTHER PLANT AUXILIARIES
>50.

Figure 16. Interactive Program, Logic Routine Example

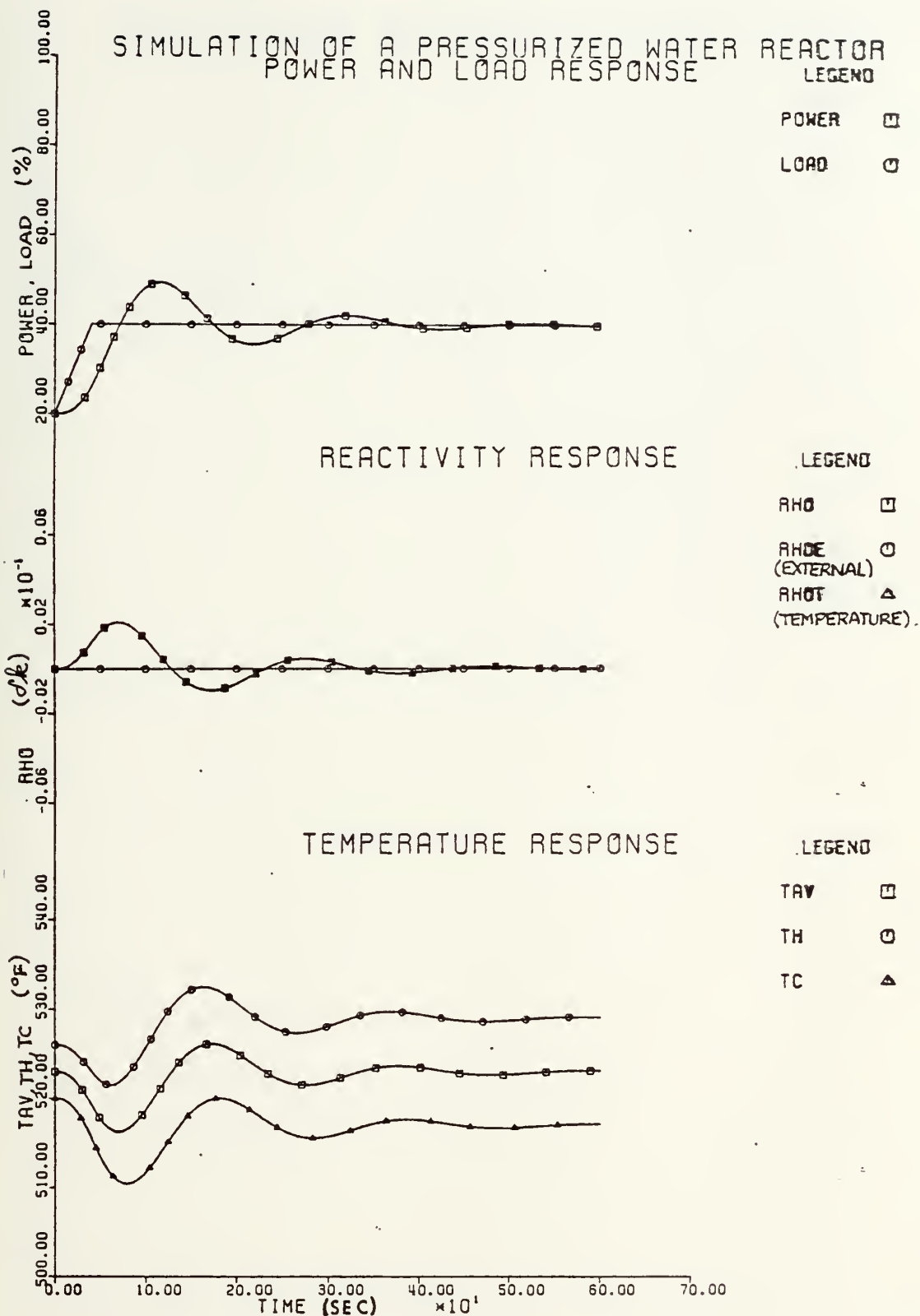


Figure 17. Power, Reactivity, and Temperature Response to a 20 to 40% Turbine Load Change

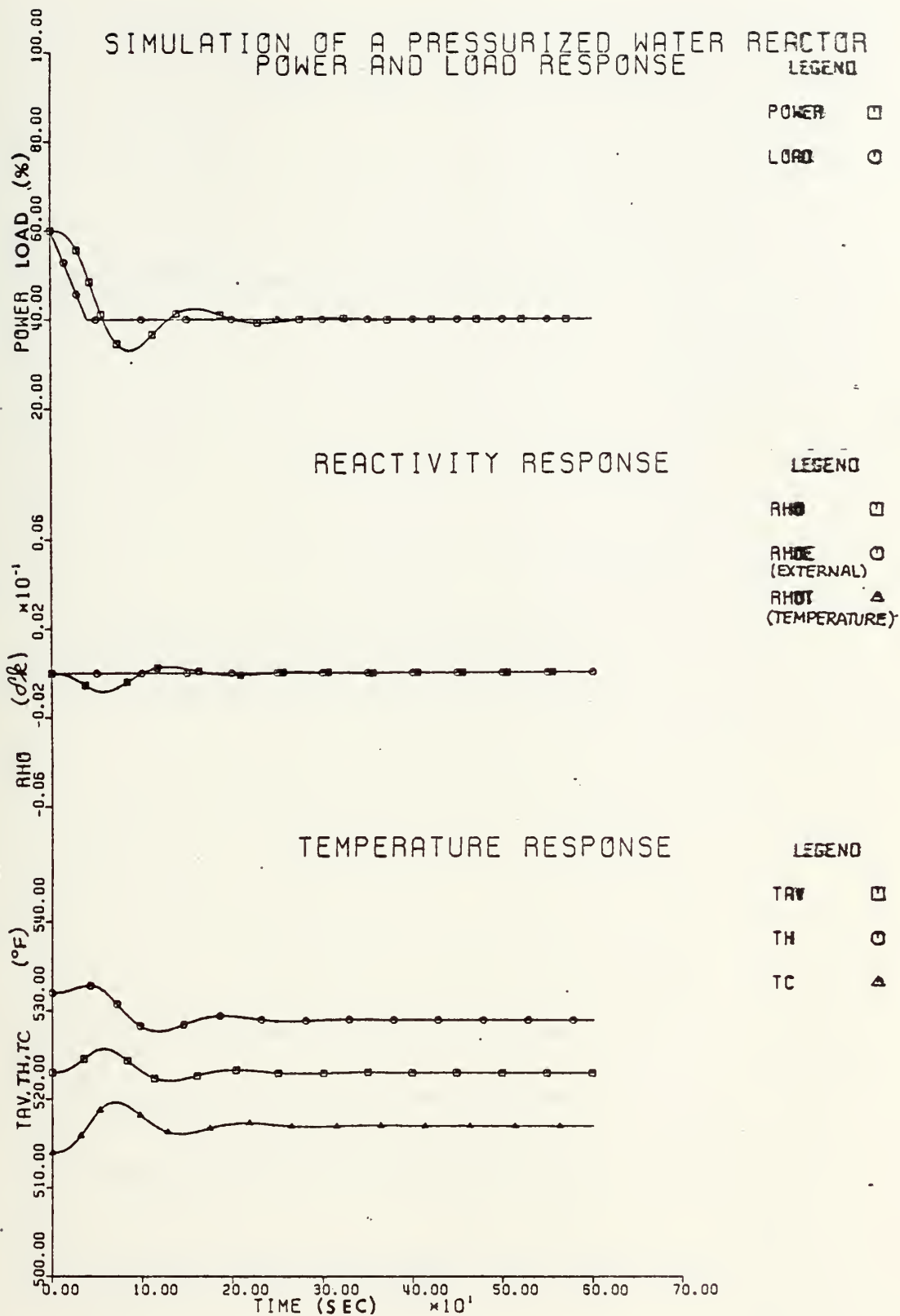


Figure 18. Power, Reactivity, and Temperature Response to a 60 to 40% Turbine Load Change

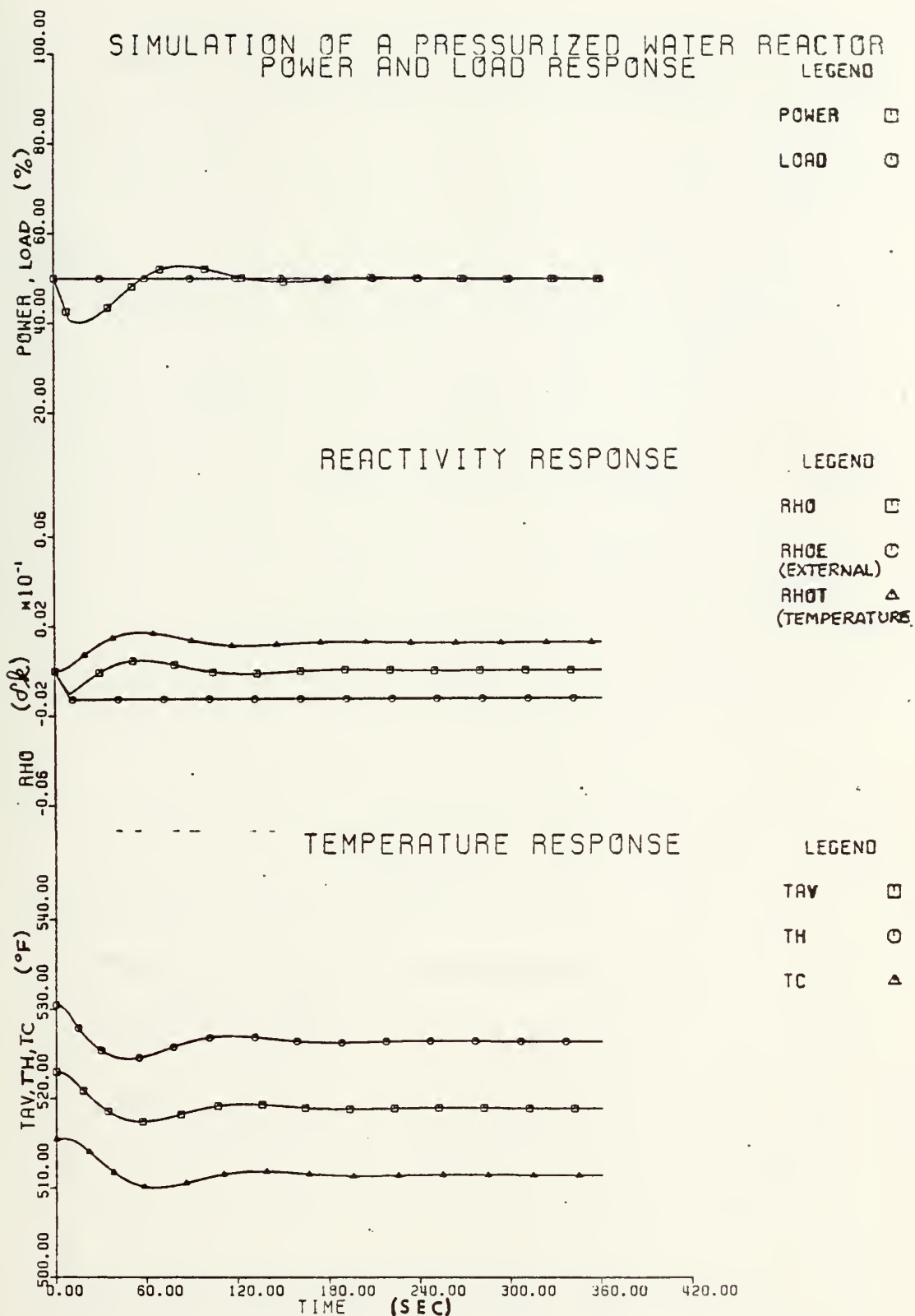


Figure 19. Power, Reactivity, and Temperature Response to a 10 second Inward Control Rod Movement

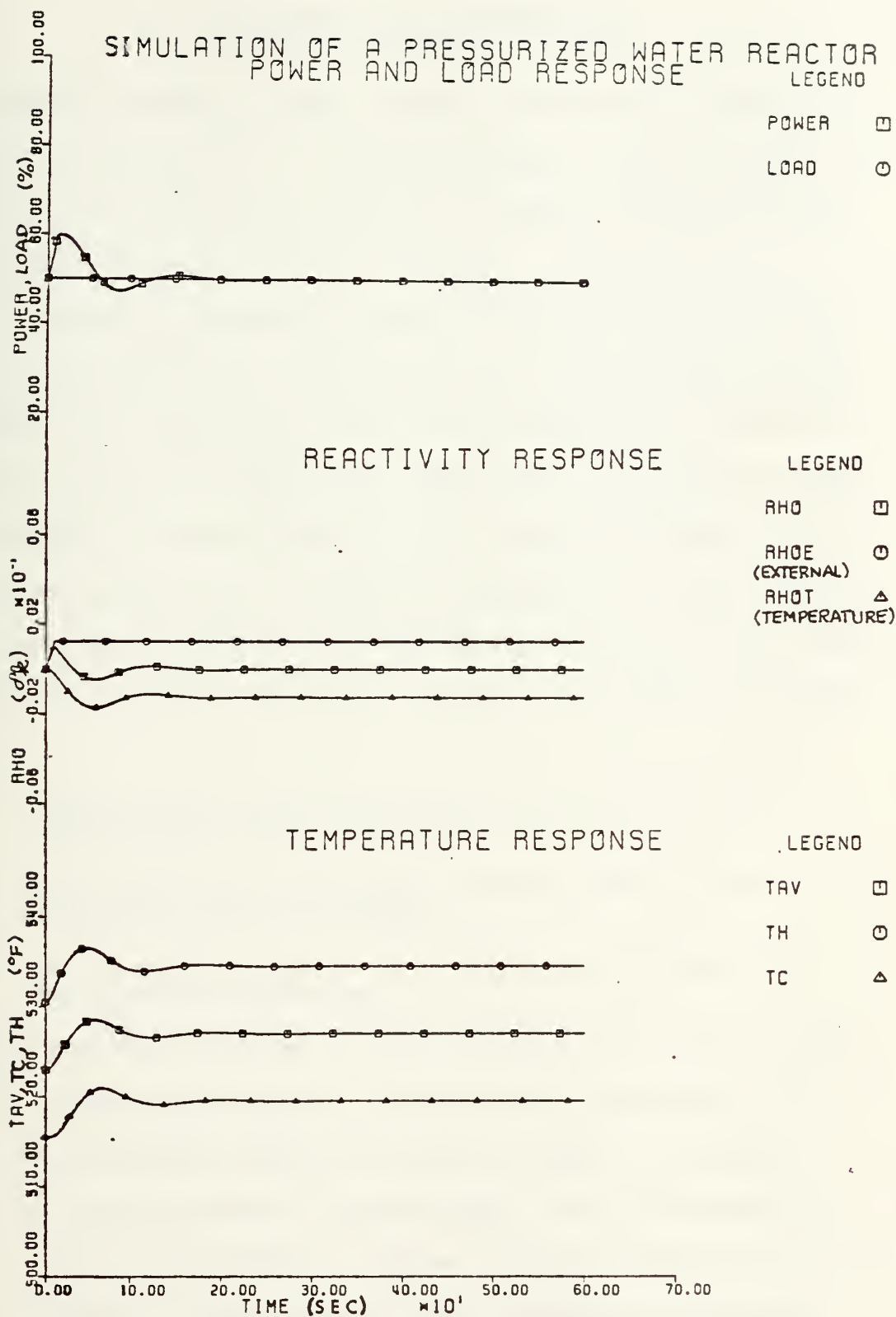


Figure 20. Power, Reactivity, and Temperature Response to a 10 second Outward Control Rod Movement

V. CONCLUSIONS AND RECOMMENDATIONS

The model responds to normal operating transients with the characteristics of a PWR plant. The interactive program provides the user with the facility to initiate the simulation or examine a real time display of post-shutdown Xenon-135 buildup and decay.

As previously discussed, the purpose of this work was to develop a learning device for nuclear engineering students at the Naval Postgraduate School. The result was a relatively simple model of a PWR power plant and a simulation program that suffers from a long execution time.

The model's simplicity resulted mainly from the consideration of an effective single group of delayed neutron precursors in the reactor kinetic equations and the use of lumped parameter analysis in the plant's heat transfer processes. The model's sophistication could be increased by:

1. Expanding the reactor kinetic equations to consider six delayed neutron precursor groups.
2. Developing a multi-section heat transfer model for the reactor and the heat exchanger.
3. Incorporating variable reactor coolant mass flow rate and heat transfer coefficients.

These additional features will also increase the complexity of the simulation program, thus aggravating the long run time problem. This concern might be eased by taking a different approach to the model's formulation than the transfer function method. While this method was readily coded using the CSMP-III language, physical insight to the plant dynamic processes was lost when the Laplace transform and subsequent grouping of constant terms was performed. The use of state variable

theory might not only allow this physical appreciation to be retained but also result in shorter program run times.

Regardless of the formulation method, a real time response to the simulation can probably only be achieved by using an analog computer. The wide range of time constants associated with the equations (prompt neutron life-times on the order of 10^{-5} seconds to Xenon decay half lives on the order of hours) require a small numerical integration time interval over a large time period. The result is a prohibitively long run time on a digital computer for an interactive program.

It is hoped that refinement of the model will be continued. While the existing model does reflect the general characteristics of a PWR plant to normal operating transients, it does exhibit relatively large power over/under shoots in response to turbine load changes as seen in Figures 17 and 18.

APPENDIX A

DEVELOPMENT OF THE XENON FEEDBACK TRANSFER FUNCTION COMPONENTS

1. Derivation of $G_x(s)$

The previously developed rate equations for ^{135}I and ^{135}Xe from the effective ^{135}Xe decay scheme shown in Figure 5 are:

$$\frac{dI}{dt} = \gamma_I \Sigma_f \phi(t) - \lambda_I I(t) \quad (\text{A } 1)$$

$$\frac{dX}{dt} = \gamma_X \Sigma_f \phi(t) + \lambda_I I(t) - \lambda_X X(t) - \sigma_a^X \phi(t) X(t) \quad (\text{A } 2)$$

where $I(t) = ^{135}\text{I}$ number density (atoms/cm³)

$X(t) = ^{135}\text{Xe}$ number density (atoms/cm³)

Σ_f = Macroscopic fission cross section of ^{235}U (cm⁻¹)

ϕ = Average integrated one group flux (neutrons/cm²sec)

γ_I = Effective ^{135}I fission yield

γ_X = Effective ^{135}Xe fission yield

λ_I = ^{135}I decay constant (sec⁻¹)

λ_X = ^{135}Xe decay constant (sec⁻¹)

σ_a^X = ^{135}Xe thermal neutron absorption cross section (cm²)

Let $I(t) = I_0 + \delta I(t)$

$X(t) = X_0 + \delta X(t)$ (A 3)

$\phi(t) = \phi_0 + \delta \phi(t)$

where the zero subscript denotes an initial steady state value and the delta prefix denotes a small perturbation about this value.

With the reactor at an initial steady state with a flux level ϕ_0 , the equilibrium values of ^{135}I and ^{135}Xe are found by setting their time dependence in equations (A 1) and (A 2) equal to zero. Thus,

$$I_0 = \frac{\gamma_I \Sigma_f \phi_0}{\lambda_I} \quad (\text{A } 4)$$

$$X_0 = \frac{\gamma_X \Sigma_f \phi_0 + \lambda_I I_0}{\lambda_X + \sigma_a^X \phi_0} \quad (\text{A } 5)$$

Substituting equation (A 4) into equation (A 5)

$$X_0 = \frac{(\gamma_X + \gamma_I) \Sigma_f \phi_0}{\lambda_X + \sigma_a^X \phi_0} \quad (\text{A } 6)$$

Substituting equations (A 3), (A 4), and (A 6) into equation (A 1) and (A 2), neglecting the $\phi \dot{X}$ term (first order approximation)

$$\frac{d}{dt} \delta I = \gamma_I \Sigma_f \delta \phi - \lambda_I \delta I \quad (\text{A } 7)$$

$$\frac{d}{dt} \delta X = \lambda_I \delta I + (\gamma_X \Sigma_f - \sigma_a^X X_0) \delta \phi - (\lambda_X + \sigma_a^X \phi_0) \delta X \quad (\text{A } 8)$$

Taking the Laplace transform of equations (A 7) and (A 8) and solving for $\delta I(s)$ and $\delta X(s)$

$$\delta I(s) = \frac{\gamma_I \Sigma_f \delta \phi(s)}{s + \lambda_I} \quad (\text{A } 9)$$

$$\delta X(s) = \frac{\lambda_I \delta I(s) + (\gamma_X \Sigma_f - \sigma_a^X X_0) \delta \phi(s)}{(s + \lambda_X + \sigma_a^X \phi_0)} \quad (\text{A } 10)$$

Substituting equation (A 9) into equation (A10)

$$\frac{d^2X(s)}{ds^2} = \frac{\lambda_I \gamma_I \Sigma_f \frac{d\phi(s)}{ds} + (s + \lambda_I)(\gamma_X \Sigma_f - \sigma_a^X X_0) \frac{d\phi(s)}{ds}}{(s + \lambda_I)(s + \lambda_X + \sigma_a^X \phi_0)}$$

Expanding, collecting terms, and solving for $\frac{d^2X(s)}{ds^2}$

$$\frac{d^2X(s)}{ds^2} = \frac{(\gamma_X \Sigma_f - \sigma_a^X X_0)s + \lambda_I(\gamma_I \Sigma_f + \gamma_X \Sigma_f - \sigma_a^X X_0)}{s^2 + (\sigma_a^X \phi_0 + \lambda_X + \lambda_I)s + \lambda_I(\sigma_a^X \phi_0 + \lambda_X)} = G_X(s)$$

This is equation (16) on page 23.

2. Derivation of α_X

The effective core neutron multiplication factor k is given by the familiar "six factor formula" found in the literature

$$k = \eta \epsilon p f P_{FNL} P_{TNL} \quad (A 11)$$

where η = Thermal fission factor of the fuel

ϵ = Fast fission factor of the fuel

p = Resonance escape probability

f = Thermal utilization factor in the core

P_{FNL} = Fast neutron non-leakage probability

P_{TNL} = Thermal neutron non-leakage probability

The core reactivity is defined as

$$\rho = \frac{k-1}{k} \quad (A 12)$$

Consider a reactor at an initial critical steady state condition with an equilibrium ^{135}Xe concentration X_0 . By the definition of criticality

$$\begin{aligned} k_0 &= 1 \\ \rho_0 &= 0 \end{aligned} \quad (\text{A } 13)$$

In this condition

$$f_0 = \frac{\Sigma_a^F}{\Sigma_{a_0}^{\text{core}}} \approx \frac{\Sigma_a^F}{\Sigma_a^F + \Sigma_a^{X_0}} \quad (\text{A } 14)$$

where Σ_a^F = Fuel macroscopic thermal neutron absorption cross section

$\Sigma_{a_0}^{\text{core}}$ = Initial core macroscopic thermal neutron absorption cross section

$\Sigma_a^{X_0}$ = Initial ^{135}Xe macroscopic thermal neutron absorption cross section

and the macroscopic thermal neutron absorption cross sections of the moderator and other core materials has been neglected, a reasonable approximation in a highly enriched core.

Now impose a small perturbation in the ^{135}Xe concentration on the core. Neglecting the effect of the ^{135}Xe perturbation on neutron leakage, a reasonable assumption for a large reactor,

$$X = X_0 + \delta X$$

$$k = k_0 + \delta k$$

$$f = f_0 + \delta f$$

$$\rho = \rho_0 + \delta \rho = \delta \rho \quad \text{since } \rho_0 = 0$$

The remainder of the terms in equation (A 11) are unchanged, thus

$$\delta k = \delta f$$

and

$$\frac{k}{k_0} = \frac{f}{f_0} = k \quad \text{since } k_0 = 1$$

where

$$f = \frac{\sum_a^F}{\sum_a^{\text{core}}} = \frac{\sum_a^F}{\sum_{a_0}^{\text{core}} + \sigma_a^x d'X} \quad (\text{A } 15)$$

from equation (A 12) and (A 13)

$$d\rho = d\rho_x = \frac{k-1}{k} = 1 - \frac{1}{k} \quad (\text{A } 16)$$

substituting equations (A 14) and (A 15) into equation (A 16)

$$d\rho_x = 1 - \frac{\sum_{a_0}^{\text{core}} + \sigma_a^x d'X}{\sum_{a_0}^{\text{core}}} = - \frac{\sigma_a^x d'X}{\sum_{a_0}^{\text{core}}} = - \frac{\sigma_a^x d'X}{\sigma_a^x X_0 + \sum_a^F}$$

or

$$\frac{d\rho_x}{d'X} = - \frac{1}{X_0 + \frac{\sigma_a^u}{\sigma_a^x} U} = \alpha_x$$

This is equation (17) on page 23. Since fuel depletion effects are not considered during the short simulated transients, the fuel number density U is assumed constant and α_x is invariant in the time and the Laplace domains.

APPENDIX B

DEVELOPMENT OF THE REACTOR HEAT TRANSFER FUNCTIONS

The previously developed equations describing heat transfer from the reactor to the coolant are

$$T_F(t) + \tau_1 \frac{dT_F(t)}{dt} = T_{AV}(t) + \frac{\tau_1}{C_F} P(t) \quad (B\ 1)$$

$$T_{AV}(t) \left[1 + 2 \frac{\tau_2}{\tau_0} \right] + \tau_2 \frac{dT_{AV}(t)}{dt} = T_F(t) + 2 \frac{\tau_2}{\tau_0} T_{Ci}(t) \quad (B\ 2)$$

$$T_{AV}(t) = \frac{1}{2} [T_{Ho}(t) + T_{Ci}(t)] \quad (B\ 3)$$

where $T_F(t)$ = Average core material temperature ($^{\circ}F$)

$T_{AV}(t)$ = Average reactor coolant temperature ($^{\circ}F$)

$T_{Ho}(t)$ = Reactor coolant outlet temperature ($^{\circ}F$)

$T_{Ci}(t)$ = Reactor coolant inlet temperature ($^{\circ}F$)

$P(t)$ = Total power generated in the core (Btu/sec)

C_F = Total thermal capacity of core materials (Btu/ $^{\circ}F$)

C_M = Total thermal capacity of coolant in the core (Btu/ $^{\circ}F$)

C = Specific heat of coolant (Btu/lbm $^{\circ}F$)

\dot{m}_m = Coolant mass flow rate (lbm/sec)

h_{FM} = Total heat transfer coefficient (Btu/sec $^{\circ}F$)

and

$$\tau_0 = \frac{C_M}{\dot{m}_m C}$$

$$\tau_1 = \frac{C_F}{h_{FM}}$$

$$\tau_2 = \frac{C_M}{h_{FM}}$$

$$\begin{aligned}
\text{Let } P(t) &= P_0 + \delta P(t) \\
T_F(t) &= T_{f0} + \delta T_F(t) \\
T_{AV}(t) &= T_{AV} + \delta T_{AV}(t) \\
T_{Ho}(t) &= T_{Ho_0} + \delta T_{Ho}(t) \\
T_{Ci}(t) &= T_{Ci_0} + \delta T_{Ci}(t)
\end{aligned} \tag{B 4}$$

where the lowest zero subscript denotes a steady state value and the delta prefix a small perturbation about this value.

From substitution of equations (B 4) into equations (B 1), (B 2), and (B 3), with the reactor in a steady state at $t \leq 0$,

$$\begin{aligned}
T_{F0} &= T_{AV0} + \frac{\bar{T}_1}{C_F} P_0 \\
T_{AV0} \left(1 + 2 \frac{\bar{T}_2}{T_0} \right) &= T_{F0} + 2 \frac{\bar{T}_2}{T_0} T_{Ci0} \\
T_{AV0} &= \frac{1}{2} (T_{Ho0} + T_{Ci0})
\end{aligned} \tag{B 5}$$

and

$$\begin{aligned}
\delta P(0) &= 0 \\
\delta T_F(0) &= \delta T_{AV}(0) = \delta T_{Ho}(0) = \delta T_{Ci}(0) = 0
\end{aligned} \tag{B 6}$$

Impose the perturbations at $t=0$. Substituting equations (B 4) and (B 5) into equations (B 1), (B 2), and (B 3) with the initial conditions given in equations (B 6),

$$\delta T_F(t) + \bar{T}_1 \frac{d}{dt} \delta T_F(t) = \delta T_{AV}(t) + \frac{\bar{T}_1}{C_F} \delta P(t) \tag{B 7}$$

$$\delta T_{AV}(t) \left[1 + 2 \frac{\bar{T}_2}{T_0} \right] + \bar{T}_2 \frac{d}{dt} [\delta T_{AV}(t)] = \delta T_F(t) + 2 \frac{\bar{T}_2}{T_0} \delta T_{Ci}(t) \tag{B 8}$$

$$\delta T_{AV}(t) = \frac{1}{2} [\delta T_{Ho}(t) + \delta T_{Ci}(t)] \tag{B 9}$$

Taking the Laplace transform of equations (B 7), (B 8), and (B 9) with the initial conditions given by equations (B 6)

$$T_F(s) [s\tau_1 + 1] = T_{AV}(s) + \frac{\tau_1}{C_F} P(s) \quad (B 10)$$

$$T_{AV}(s) [s\tau_2 + 1 + 2 \frac{\tau_2}{\tau_0}] = T_F(s) + 2 \frac{\tau_2}{\tau_0} T_C(s) \quad (B 11)$$

$$T_{AV}(s) = \frac{1}{2} [T_H(s) + T_C(s)] \quad (B 12)$$

where the delta prefix and lowest subscript have been deleted for readability

Substituting equation (B 12) into equations (B 10) and (B 11)

$$T_F(s\tau_1 + 1) = \frac{1}{2} (T_H + T_C) + \frac{\tau_1}{C_F} P \quad (B 13)$$

$$\frac{1}{2} (T_H + T_C) (s\tau_2 + 1 + 2 \frac{\tau_2}{\tau_0}) = T_F + 2 \frac{\tau_2}{\tau_0} T_C \quad (B 14)$$

where the s domain dependence notation has been deleted for readability.

Solving equation (B 13) for T_F and substituting into equation (B 14)

$$\frac{1}{2} (T_H + T_C) (s\tau_2 + 1 + 2 \frac{\tau_2}{\tau_0}) (s\tau_1 + 1) = \frac{1}{2} (T_H + T_C) + \frac{\tau_1}{C_F} P + 2 \frac{\tau_2}{\tau_0} (s\tau_1 + 1) T_C$$

After expanding the products and collecting terms

$$T_H \left[s^2 \frac{\tau_1 \tau_2}{2} + \left(\frac{\tau_2}{2} + \frac{\tau_1}{2} + \frac{\tau_1 \tau_2}{\tau_0} \right) s + \frac{\tau_2}{\tau_0} \right] = -T_C \left[s^2 \frac{\tau_1 \tau_2}{2} + \left(\frac{\tau_2}{2} + \frac{\tau_1}{2} - \frac{\tau_1 \tau_2}{\tau_0} \right) s - \frac{\tau_2}{\tau_0} \right] + \frac{\tau_1}{C_F} P$$

After multiplying through by and solving for $T_H(s)$

$$\mathcal{L}T_{H_0}(s) = \frac{- \left\{ \frac{\tau_1 \tau_2}{2} s^2 + \left[\frac{\tau_0}{\tau_2} \left(\frac{\tau_1}{\tau_2} + 1 \right) - \tau_1 \right] s - 1 \right\} \mathcal{L}T_{C_1}(s) + \gamma \mathcal{L}P(s)}{\left\{ \frac{\tau_1 \tau_2}{2} s^2 + \left[\frac{\tau_0}{\tau_2} \left(\frac{\tau_1}{\tau_2} + 1 \right) + \tau_1 \right] s + 1 \right\}}$$

where $\gamma = \frac{\tilde{I}_1 \tilde{I}_0}{c_p \tilde{I}_2}$ and all notation has been restored.

This is equation (26) on page 28.

APPENDIX C

DEVELOPMENT OF THE HEAT EXCHANGER TRANSFER FUNCTION

The previously derived heat exchanger heat transfer equations are

$$T_{Co}(t) = \frac{2 T_S(t) - T_{Hi}(t) [1 - K_1]}{[1 + K_1]} \quad (C 1)$$

$$\frac{C_S}{h_{TM}} \frac{dT_S(t)}{dt} + T_S(t) = \frac{1}{2} [T_{Hi}(t) + T_{Co}(t)] + \frac{1}{h_{TM}} P_L(t) \quad (C 2)$$

where $T_{Hi}(t)$ = Heat exchanger coolant inlet ($^{\circ}F$)

$T_{Co}(t)$ = Heat exchanger coolant outlet ($^{\circ}F$)

$T_S(t)$ = Saturated steam pressure (PSIQ)

$P_L(t)$ = Power delivered by heat exchanger (Btu/sec)

C_S = Total thermal capacity of heat exchanger metal and secondary water and steam (Btu/ $^{\circ}F$)

C = Specific heat of coolant (Btu/lbm $^{\circ}F$)

h_{TM} = Total heat transfer coefficient (Btu/ $^{\circ}F$.sec)

\dot{m}_m = Coolant mass flow rate (lbm/sec)

and

$$K_1 = \frac{2 \dot{m}_m C}{h_{TM}}$$

Let $T_{Hi}(t) = T_{Hi_0} + \delta T_{Hi}(t)$

$$T_{Co}(t) = T_C + \delta T_{Ho}(t) \quad (C 3)$$

$$T_S(t) = T_S + \delta T_S(t)$$

$$P_L(t) = P_{L_0} + \delta P_L(t)$$

where the lowest zero subscript denotes a steady state value and the delta prefix a small perturbation about this value.

With the plant in an initial steady state at $t \leq 0$, equations (C 1) and (C 2) are

$$T_{c_o} = \frac{2T_{s_o} - T_{H_{i_o}} [1 + K_1]}{[1 + K_1]} \quad (C 4)$$

$$T_{s_o} = \frac{1}{2} (T_{H_{i_o}} + T_{c_o}) + \frac{1}{h_{TM}} P_{L_o} \quad (C 5)$$

and

$$\delta P_L(0) = 0 \quad (C 6)$$

$$\delta T_{H_i}(0) = \delta T_{c_o}(0) = \delta T_s(0) = 0$$

Impose the perturbations substituting equations (C 3), (C 4), and (C 5) into equations (C 1) and (C 2)

$$\delta T_{c_o} = \frac{2\delta T_s - \delta T_{H_i} (1 - K_1)}{(1 + K_1)} \quad (C 7)$$

$$\frac{C_s}{h_{TM}} \frac{d}{dt} \delta T_s(t) + \delta T_s(t) = \frac{1}{2} [\delta T_{H_i}(t) + \delta T_{c_o}(t)] - \frac{1}{h_{TM}} \delta P_L(t) \quad (C 8)$$

Taking the Laplace transforms of equations (C 7) and (C 8) with the initial conditions given by equation (C 6)

$$\delta T_s(s) = \frac{\frac{1}{2} [\delta T_{c_o}(s) + \delta T_{H_i}(s)] - \frac{1}{h_{TM}} \delta P_L(s)}{\tau_s s + 1}$$

where $\tau_s = \frac{C_s}{h_{TM}}$

$$\delta T_{c_o}(s) = \left\{ 2\delta T_s(s) - \delta T_{H_i}(s) [1 - K_1] \right\} / [1 + K_1]$$

These are equations (34) and (35) on page 33.

APPENDIX D

DEVELOPMENT OF THE PRIMARY PIPING TRANSFER FUNCTIONS

1. Transport Delay Transfer Functions

The transport delay of the coolant between the outlet of the reactor and the heat exchanger inlet plenum can be expressed as

$$T_{Hi_p}(t) = T_{Ho}(t - \tau_3) \quad (D 1)$$

where $T_{Hi_p}(t)$ = Heat exchanger inlet plenum coolant temperature ($^{\circ}F$)

$T_{Ho}(t)$ = Reactor coolant outlet temperature ($^{\circ}F$)

τ_3 = Transport time delay (sec)

Rearranging terms and expanding equation (D 1) in a Taylor series

$$T_{Ho}(t) = T_{Hi_p}(t) + \tau_3 \frac{dT_{Hi_p}(t)}{dt} + \frac{\tau_3^2}{2!} \frac{d^2 T_{Hi_p}(t)}{dt^2} + \dots \quad (D 2)$$

For slow temperature changes, the second order and higher terms in equation (D 2) may be ignored.

$$T_{Hi_p}(t) + \tau_3 \frac{dT_{Hi_p}(t)}{dt} = T_{Ho}(t) \quad (D 3)$$

Let

$$T_{Hi}(t) = T_{Hi_{po}} + \delta T_{Hi_p}(t) \quad (D 4)$$

$$T_{Ho}(t) = T_{Ho_{oo}} + \delta T_{Ho}(t)$$

where the lowest zero subscript denotes a steady state value and the delta prefix a small perturbation about this value.

From substitution of equations (D 4) into equation (D 3) and for the temperatures in a steady state at $t \leq 0$

$$T_{Hi_{po}} = T_{Ho_0} \quad (D 5)$$

and

$$\delta T_{Hi_p}(t) = \delta T_{Ho}(t) = 0 \quad (D 6)$$

Impose the perturbations at $t=0$. Substituting equations (D 4) and (D 5) into equation (D 3) with the initial conditions given in equation (D 6)

$$\delta T_{Hi_p}(t) + \tau_3 \frac{d}{dt} \delta T_{Hi_p}(t) = \delta T_{Ho}(t) \quad (D 7)$$

Taking the Laplace transform of equation (D 7) and solving for $\delta T_{Hi} / \delta T_{Ho}$

$$\frac{\delta T_{Hi_p}(s)}{\delta T_{Ho}(s)} = \frac{1}{1 + \tau_3 s}$$

This is equation (39) on page 36.

By following a similar development, the transport delay between a perturbation in the heat exchanger coolant outlet temperature and the resulting perturbation in the reactor inlet plenum coolant temperature can be expressed by the transfer function

$$\frac{\delta T_{ci_p}(s)}{\delta T_{co}(s)} = \frac{1}{1 + \tau_4 s}$$

where δT_{ci_p} = Reactor inlet plenum coolant temperature perturbation ($^{\circ}F$)

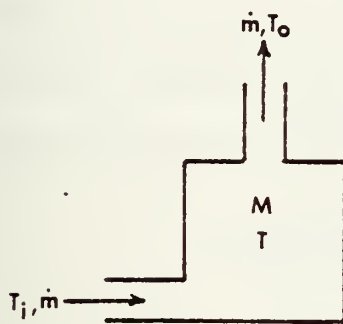
δT_{co} = Heat exchanger coolant outlet temperature perturbation ($^{\circ}F$)

τ_4 = Transport time delay (sec)

This is equation (40) on page

2. Mixing Delay Transfer Function

Let the reactor and heat exchanger coolant inlet plenums be represented by the control volume shown in Figure D-1.



M = Mass of coolant in plenum (lbm/sec)

T = Average plenum coolant temperature ($^{\circ}\text{F}$)

T_i = Plenum inlet coolant temperature ($^{\circ}\text{F}$)

T_o = Plenum outlet coolant temperature ($^{\circ}\text{F}$)

m = Coolant mass flow rate (lbm/sec)

Figure D-1. Mixing Volume

If perfect mixing is assumed, the coolant outlet temperature is equal to the average coolant temperature in the plenum. At heat balance on the volume requires that the net heat flowing into the plenum be equal to the increase in stored energy. Assuming no ambient heat losses, and assuming the temperature of the plenum structural material remains constant, the heat balance per unit time over a time interval Δt is

$$\dot{m} \Delta t C [T_i(t) - T_o(t)] = MC \frac{dT(t)}{dt} \Delta t = \dot{m} \Delta t C \frac{dT_o(t)}{dt}$$

where C = specific heat of coolant

The resulting equation is

$$\frac{M}{\dot{m}} \frac{dT_o(t)}{dt} + T_o(t) = T_i(t) \quad (\text{D } 8)$$

where $T = \frac{M}{\dot{m}}$ = mixing time delay.

Let

$$T_i(t) = T_{i0} + \delta T_i(t) \quad (D\ 9)$$

$$T_o(t) = T_{o0} + \delta T_o(t)$$

where the lowest zero subscript denotes an initial steady state value and the delta prefix a small perturbation about this value.

Substituting equations (D 8) into equation (D 9)

$$\tau \frac{d}{dt} \delta T_o(t) + \delta T_o(t) = \delta T_i(t) \quad (D10)$$

Taking the Laplace transform of equation (D10) and solving for $\delta T_o / \delta T_i$

$$\frac{\delta T_o(s)}{\delta T_i(s)} = \frac{1}{1 + \tau s}$$

This is equation (41) on page 37.

INTERACTIVE PROGRAM LISTING

```

C      INTEGER T
C      DIMENSION X(3000), TIME(3000), TIMEH(3000), LAB1(29), LAB2(21),
C      $LAB3(10)
C      GRAPH TITLE: 3 0 S T S H U T D O W N X E N O
C      DATA LAB1 / 80,79,83,84,32,83,72,85,84,68,79,87,78,32,88,69,78,79,
C      *78,32,84,82,65,78,83,73,59,78,84 /
C      SCALE TITLE: K E V D N I 3 5 A T O M S / C M 3
C      DATA LAB2 / 88,69,78,79,78,32,49,51,53,32,65,84,79,77,83,47,67,77,
C      *51 /
C      SCALE TITLE: T I M E ( H R S )
C      DATA LAB3 / 84,73,77,69,32,40,72,82,83,41 /
C      INITIALIZE FOLLOWING PLANT PARAMETERS: (ATOMS/CM3)
C      URANIUM-235 NUMBER DENSITY (ATOMS/CM3)
C      URANIUM-235 MICROSCOPIC ABSORPTION CROSS SECTION (CM2)
C      URANIUM-235 MICROSCOPIC FISSION CROSS SECTION (CM2)
C      URANIUM-235 MICROSCOPIC ABSORPTION CROSS SECTION (CM2)
C      XENON-135 MICROSCOPIC ABSORPTION CROSS SECTION (CM2)
C      XENON-135 MICROSCOPIC FISSION YIELD OF IODINE-135 (%)
C      EFFECTIVE FISSION YIELD OF XENON-135 (%)
C      IODINE-135 DECAY CONSTANT (1/SEC)
C      XENON-135 DECAY CONSTANT (1/SEC)
C      REACTOR COOLANT MASS FLOW RATE (LBM/SEC)
C      REACTOR COOLANT SPECIFIC THERMAL CAPACITY (BTU/LBM F)
C      STEAM GENERATOR TOTAL EFFECTIVE HEAT TRANSFER
C      COEFFICIENT (BTU/SEC F)
C      REAL IJ,KI,LAMDAL/2.875E-5/,LAMDAX/2.0917E-5/,MDOTM/6280./,LOADJ,
C      1LOADOM,LJADE
C      U235 = 6.07E+15
C      SIGMAA = 393.E-24
C      SIGMAF = 334.E-24
C      SIGMAX = 1.5625E-18
C      GAMMAX = 0.06396
C      GAMMAI = 0.00228
C      CPM = 1.16
C      HTM = 5919.
C      PRINT PROGRAM INTRODUCTION
C      WRITE(5,6000)
C      READ INITIAL STEADY STATE POWER LEVEL (PWRO)
C      WRITE (6,6002) PWRO
C      READ (5,5000) PWRO

```



```

CC C DETERMINE IF INCORRECT DATA ENTERED
C C IF ( PWRD .GE. 10. ) AND( PWRD .LE. 100. ) GO TO 20
C C WRITE NOTICE OF INCORRECT DATA ENTRY
C C WRITE (6,6200)
C C GO TO 15
C C
C C TRANSFER PWRD TO INITIAL DATA FILE
C C WRITE (1,1000) PWRD
C C
C C CALCULATE FOLLOWING INITIAL STEADY STATE PARAMETERS:
C C PWRDM INITIAL REACTOR POWER (MW)
C C FLUXO ONE GROUP NEUTRON FLUX LEVEL (NEUTONS/CM2 SEC)
C C XJ INITIAL XENON-135 CONCENTRATION (ATOMS/CM3)
C C RJOXO INITIAL XENON-135 REACTIVITY (DELTA K)
C C TMRJ INITIAL AVERAGE REACTOR COOLANT TEMPERATURE (F)
C C THJD INITIAL FUEL TEMPERATURE (F)
C C TCJJ INITIAL REACTOR COOLANT JETLET TEMPERATURE (F)
C C TCJJ INITIAL REACTOR COOLANT INLET TEMPERATURE (F)
C C TCJJ INITIAL REACTOR COOLANT OUTLET TEMPERATURE (F)
C C TCJJ INITIAL STEAM GENERATOR COOLANT INLET TEMPERATURE (F)
C C TSQ INITIAL STEAM GENERATOR COOLANT OUTLET TEMPERATURE (F)
C C PSD INITIAL STEAM PRESSURE (PSIA)
C C LOADDO INITIAL TURBINE LOAD (%)
C C LOADOM INITIAL TURBINE LOAD (MWE)
C C
C C PWROM = PWRD * 2.31
C C FLUXJ = 2.E+12 * PWRD
C C XO = ( GAMMAI + GAMMAX ) * U235 * SIGMAF * FLJXD / ( LAMDAX + SIGMAX * FLUXO )
C C TMRD = 523. + 3.77 * PWRD
C C TFO = TMRD + .15 * PWRD
C C THJD = TMRD - .15 * PWRD
C C TCIO = THOO
C C TCNO = TCIO
C C KI = 2. * MDDOT * CM / HTM
C C TSO = ( THIO * (1.-KI) ) + TCJJ * (1.+KI) ) * 0.5
C C PSQ = EXP { (A.JG(TSD)) - 4.77123 / 0.22151 }
C C LOADDO = PWRD
C C LOADOM = LOADJ * .68
C C
C C PRINT INITIAL STEADY STATE PARAMETERS
C C WRITE(5,6301) PWRD, PWROM, FLUXJ, XO, TFO, TMRD, THOO, TCIO, TSO, PSD, LOADDO,
C C $, LOADOM

```

၀
ဗဗ ဗ ဗဏဗဗဗဗဗဗဗဗဗဗဗဗဗဗဗဗဗဗဗဗဗဗဗ

UU

C	25	DETERMINE DESIRED SIMULATION	INAO1030
C		WRITE (6,6003) ANSWER	INAO1040
C		READ (5,5000) ANSWER	INAO1050
C		DETERMINE IF INCORRECT DATA ENTERED	INAO1060
C		IF (IANS = IFIX(ANSWER)) GO TO 33	INAO1080
C		IF (IANS.EQ.1) GO TO 33	INAO1090
C		IF (IANS.EQ.2) GO TO 80	INAO1100
C		PRINT NOTICE OF INCORRECT DATA ENTRY	INAO1120
C		WRITE (5,5200)	INAO1130
C		GO TO 25	INAO1140
C	30	DETERMINE INITIAL REACTIVITY CHANGE MECHANISM	INAO1150
C		WRITE (6,6004)	INAO1160
C		READ (5,5000) ANSWER	INAO1170
C		DETERMINE IF INCORRECT DATA ENTERED	INAO1180
C		IF (IANS = IFIX(ANSWER)) GO TO 35	INAO1190
C		IF (IANS.EQ.1) GO TO 35	INAO1200
C		IF (IANS.EQ.2) GO TO 65	INAO1210
C		PRINT NOTICE OF INCORRECT DATA ENTRY	INAO1220
C		WRITE (5,5200)	INAO1230
C		GO TO 33	INAO1240
C		READ TIME LENGTH (SEC) OF BANK CONTROL ROD MOVEMENT (TSHIM)	INAO1250
C	35	WRITE (6,6005) TSHIM	INAO1260
C		READ (5,5000) TSHIM	INAO1270
C		DETERMINE IF INCORRECT DATA ENTERED	INAO1280
C		IF (TSHIM.GE.0 .AND. TSHIM.LE.10.) GO TO 40	INAO1300
C		PRINT NOTICE OF INCORRECT DATA ENTRY	INAO1310
C		WRITE (6,6200)	INAO1320
C		GO TO 35	INAO1330
C	40	TRANSFER TSHIM TO INITIAL DATA FILE	INAO1340
C		WRITE (1,1000) TSHIM	INAO1350
C	45	DETERMINE THE DIRECTION OF BANK CONTROL ROD MOVEMENT (DSHIM)	INAO1370
C		WRITE (6,6006) DSHIM	INAO1380
C		READ (5,5000) DSHIM	INAO1390
C		DETERMINE IF INCORRECT DATA ENTERED	INAO1400
C		IF (DSHIM.EQ.1) GO TO 55	INAO1410
C		IF (DSHIM.EQ.2) GO TO 50	INAO1420
C		PRINT NOTICE OF INCORRECT DATA ENTRY	INAO1430
C		WRITE (6,6200)	INAO1440
C		GO TO 45	INAO1450
C		DETERMINE IF INCORRECT DATA ENTERED	INAO1460
C		IF (DSHIM.EQ.1) GO TO 55	INAO1470
C		IF (DSHIM.EQ.2) GO TO 50	INAO1480
C		PRINT NOTICE OF INCORRECT DATA ENTRY	INAO1490
C		WRITE (6,6200)	INAO1500
C		GO TO 45	INAO1510
C			INAO1520

C	SET SHIM DIRECTION TO INSERT POSITIVE REACTIVITY ("OUT")	INAO1530
50	DSHIM = 1.	INAO1540
C	GO TO 50	INAO1550
55	SET SHIM DIRECTION TO INSERT NEGATIVE REACTIVITY ("IN")	INAO1560
C	DSHIM = -1.	INAO1570
60	TRANSFER DSHIM TO INITIAL DATA FILE	INAO1580
C	WRITE (1,1000) DSHIM	INAO1590
C	SET FINAL TURBINE LOAD = INITIAL TURBINE LOAD	INAO1600
C	LOADF = LOAD0	INAO1610
C	TRANSFER LOADF TO INITIAL DATA FILE	INAO1620
C	WRITE (1,1000) LOADF	INAO1630
C	GO TO 200	INAO1640
65	SET THE TIME LENGTH (TSHIM) AND DIRECTION (DSHIM) OF RANK CONTROL	INAO1650
C	ROD MOVEMENT = 3	INAO1660
C	TSHIM = 0.	INAO1670
C	DSIM = 0.	INAO1680
C	TRANSFER TSHIM AND DSHIM TO THE INITIAL DATA FILE	INAO1690
C	WRITE (1,1000) TSHIM	INAO1700
C	WRITE (1,1000) DSHIM	INAO1710
70	READ FINAL TURBINE LOAD (LOADF)	INAO1720
C	WRITE (6,6007)	INAO1730
C	READ (5,5000) LOADF	INAO1740
C	DETERMINE IF INCORRECT DATA ENTERED	INAO1750
C	IF (LOADF .GE. 10. .AND. LOADF .LE. 100.) GO TO 75	INAO1760
C	PRINT NOTICE OF INCORRECT DATA ENTRY	INAO1770
C	WRITE (6,6200)	INAO1780
C	GO TO 70	INAO1790
75	TRANSFER LOADF TO INITIAL DATA FILE	INAO1800
C	WRITE (1,1000) LOADF	INAO1810
C	GO TO 200	INAO1820
80	POST SHUTDOWN XENON TRANSIENT ALGORITHM	INAO1830
C	CONTINUE	INAO1840
C	CALCULATE EQUILIBRIUM IODINE-135 AT TIME OF SHUTDOWN	INAO1850
C	IO = GAMMAI * J235 * SIGMAF * FLUX0 / LAMDAI	INAO1860
C	CONVERT LAMDAI AND LAMDA X TO 1/MIN	INAO1870
	LAMDAI = LAMDAI * 60.	INAO1880
	LAMDA X = LAMDA X * 60.	INAO1890


```

C      CALCULATE INSTANTANEOUS XENON-135 CONCENTRATION
DO 85 T=1,300,1
  TIME(T) = FLOAT(T-1)
  X1 = XJ * EXP(-LAMDAI*TIME(T))
  X2 = LAMDAI * IO/(LAMDAI-LAMDAI)
  X3 = EXP(-LAMDAI*TIME(T)) - EXP(-LAMDAI*TIME(T))
  CONVERT TIME TO HOURS
  TIME(T) = TIME(T)/60.
  X(T) = X1 + X2 * X3
  DETERMINE IF THERE IS A POST SHUTDOWN XENON PEAK
  TMAX1 = 1/(LAMDAI-LAMDAI)
  TMAX2 = LAMDAI/LAMDAI
  TMAX3 = 1. + (1.-1./TMAX2)*XJ/IO
  TMAX = TMAX1 * ALOG(TMAX2/TMAX3)
  IF (TMAX) 95,95,90
  CALCULATE THE PEAK XENON CONCENTRATION
  XMAX1 = XJ * EXP(-LAMDAI*TIME(T))
  XMAX2 = LAMDAI * IO/(LAMDAI-LAMDAI)
  XMAX3 = EXP(-LAMDAI*TIME(T)) - EXP(-LAMDAI*TIME(T))
  XMAX = XMAX1 + XMAX2 * XMAX3
  GO TO 100
XMAX = XJ
TMAX = 0.0
C      CALCULATE THE PEAK XENON REACTIVITY
ALPHA = -SIGMAX/(1.25*J235*SIGMAA)
RHOMAX = XMAX * ALPHA
TECHTRJNIX 4012 GRAPHICS ALGORITHM
L1 = 29
L2 = 21
L3 = 10
CALL INITT(120)
CALL BNPTS(300, 970)
CALL SLIMX(100, 665)
CALL SLIMY(175, 665)
CALL CHECK(TIME, X)
CALL DISPLAY(TIME, X)
CALL NCURSR(325, 695, L1, LAB1)
CALL WJVBAS(50, 710)
CALL VJABEL(L2, LAB2)
CALL NJTATE(455, 100, L3, LAB3)
CALL XZERO(0)
CALL FINIT(70, 72)
C

```

INA01970
 INA02010
 INA02020
 INA02030
 INA02040
 INA02050
 INA02060
 INA02070
 INA02080
 INA02090
 INA02100
 INA02110
 INA02120
 INA02130
 INA02140
 INA02150
 INA02160
 INA02170
 INA02180
 INA02190
 INA02200
 INA02210
 INA02220
 INA02230
 INA02240
 INA02250
 INA02260
 INA02270
 INA02280
 INA02290
 INA02300
 INA02310
 INA02320
 INA02330
 INA02340
 INA02350
 INA02360
 INA02370
 INA02380
 INA02390
 INA02400
 INA02410
 INA02420
 INA02430
 INA02440
 INA02450
 INA02460
 INA02470


```

6301  FORMAT(' ', /, ' INITIAL STEADY STATE PARAMETERS: ', /, 5X, ' REACTOR PWR INAO02970
SER=, F5.2, CM2, ' FUEL TEMPERATURE= ', F6.2, ' XENON-135 CONCENTRATION= ', F9.2, ' ATOMS/CM3, ' INAO02980
$NEUTRON FLUX= ', F9.2, ' REACTOR COOLANT INAO02990
$, /, 5X, ' MODERATOR TEMPERATURE= ', F5.2, ' F, /, 5X, ' REACTOR COOLANT OUTLET TEMPERATURE= ', F6.2, ' INAO03000
$MODERATOR TEMPERATURE= ', F6.2, ' F, /, 5X, ' REACTOR COOLANT INLET TEMPERATURE= ', F6.2, ' INAO03010
$MODERATOR TEMPERATURE= ', F6.2, ' F, /, 5X, ' REACTOR COOLANT INLET TEMPERATURE= ', F6.2, ' INAO03020
$MODERATOR TEMPERATURE= ', F6.2, ' F, /, 5X, ' REACTOR COOLANT INLET TEMPERATURE= ', F6.2, ' INAO03030
$MODERATOR TEMPERATURE= ', F6.2, ' F, /, 5X, ' REACTOR COOLANT INLET TEMPERATURE= ', F6.2, ' INAO03040
$MODERATOR TEMPERATURE= ', F6.2, ' F, /, 5X, ' REACTOR COOLANT INLET TEMPERATURE= ', F6.2, ' INAO03050
6400  $FORMAT(' ', /, 4X, ' PRE-SHUTDOWN POWER = ', F6.2, ' %, /, 10X, ' PEAK XENON T INAO03060
$TIME = ', F6.2, ' HR, ' REACTIVITY = ', F6.2, ' %, ' DELTA K, ' INAO03070
$, /, 10X, ' PEAK XENON REACTIVITY = ', F6.2, ' %, ' DELTA K, ' INAO03080
6401  $FORMAT(' ', /, 4X, ' INITIAL REACTIVITY CHANGE = ', F6.2, ' %, ' DELTA K, ' INAO03090
$FOR POWER = ', F6.2, ' %, ' DELTA K, ' INAO03100
$DIRECTION = ', F6.2, ' (1="OUT", -1="IN")', /, 5X, ' FINAL TURBINE LOAD INAO03110
6500  $FORMAT(' ', /, 4X, ' INPUTS HAVE BEEN ENTERED INTO THE SIMULATION PROG INAO03120
$RAM WHICH HAS BEEN SENT, /, ' TO THE BATCH SYSTEM FOR EXECUTION', INAO03130
STOP INAO03140
END INAO03150
      INAO03160

```


הַיְיטִי : הַיְיטִי :

```

**
**
** INITIAL
**
** ZERO COUNTERS (T1,T2) FOR THE AVERAGE REACTOR COOLANT TEMPERATURE
** (TMRT) CONTROL ALGORITHM:
**
** ZERO THE FOLLOWING FLAGS (INDICATORS) FOR THE REACTOR PROTECTION AND-
** TMRT CONTROL ALGORITHMS (THE FUNCTION OF THESE FLAGS WHEN SET IS DE-
** SCRIBED, WHEN ZEROED THEY ARE NEUTRAL):
** SDFLAG PROTECTIVE FULL INSERTION OF CONTROL RODS (SCRAM)
** CFLAG1 PROTECTIVE INSERTION OF CONTROL RODS (CUTBACK)
** CFLAG2 SCRAM ON REOCCURRING CUTBACK
** CFLAG3 ZERO CFLAG1 RATE CUTBACK CONDITION
** CFLAG4 HIGH STARTUP COOLANT OUTLET TEMPERATURE ALARM
** THFLAG HIGH REACTOR TMRT OUTSIDE NORMAL TEMPERATURE BAND OF +/- 5F
** SFLAG1 CONTROL ROD MOVEMENT TO RETURN TMRT TO NOT
** SFLAG2 ZERO SFLAG1
**
** INCON SD=LAG=0, CFLAG1=0, CFLAG2=0, CFLAG3=0, CFLAG4=0, THFLAG=0, . . .
** TFLAG=0, SFLAG1=0, SFLAG2=0, T1=0, T2=0, T=0

```

ESTABLISH STORAGE FOR DYNAMIC SECTION TRANSFER FUNCTION COEFFICIENTS:
STORAGE C(3),D(3),E(2),F(3),G(2),H(2),I(3)

ESTABLISH FOLLOWING PLANT CONSTANTS:

U235 URANIUM-235 NUCLEAR DENSITY (ATOMS/CM3)
ALPHA PROMPT NEUTRON LIFETIME (SEC)
BETA EFFECTIVE DELAYED NEUTRON PERCENTAGE (DELTA K)
GAMMA1 EFFECTIVE FISSION YIELD OF IODINE-135
GAMMA2 EFFECTIVE FISSION YIELD OF XENON-135
LAMDA1 DELAYED NEUTRON PRECURSOR DECAY CONSTANT (1/SEC)
LAMDA2 XENON-135 EFFECTIVE DECAY CONSTANT (1/SEC)
LAMDA3 XENON-135 EFFECTIVE REACTION RATE (DELTA K/SEC)
RSHIM BANK CONTROL ROD MOVEMENT (SHIM) REACTOR METAL (BTU/F)
SIGMAA URANIUM-235 MICROSCOPIC ABSORPTION CROSS SECTION (CM2)
SIGMAAF XENON-135 MICROSCOPIC ABSORPTION CROSS SECTION (CM2)
SIGMAAX TEMPERATURE COEFFICIENT OF REACTIVITY (DELTA K/F)
ALPHA1 CONTROL ROD MOVEMENT (SHIM) REACTOR METAL (BTU/F)
RSHIM TOTAL THERMAL CAPACITY OF COOLANT IN REACTOR (BTU/F)
CF TOTAL THERMAL CAPACITY OF STEAM (BTU/F)
CM TOTAL THERMAL CAPACITY (BTU/F)
CPM STEAM GENERATOR (S/G) THERMAL CAPACITY (BTU/LBM F)
HFM REACTOR TOTAL HEAT TRANSFER COEFFICIENT (BTU/SEC F)
HTM REACTOR TOTAL HEAT TRANSFER COEFFICIENT (BTU/SEC F)
MDOTM REACTOR COOLANT MASS FLOW RATE (LBM/SEC)
TAU3 REACTOR COOLANT FLOW TIME DELAY (SEC)
TAU4 HOT LEG COOLANT FLOW TIME DELAY (SEC)
TAU5 COLD LEG COOLANT FLOW TIME DELAY (SEC)
RLOAD INLET PIPE LOAD CHANGE RATE (.5%/SEC)

CONSTANT U235=5.07E+16, BETA=77.0E-4, GAMMA1=0.06386, GAMMA2=2.875E-5, GAMMA3=0.00228, LAMDA1=2.875E-5, LAMDA2=2.0917E-5, LAMDA3=2.0917E-5, SIGMAA=393.5E-24, SIGMAAF=334.5E-24, SIGMAAX=1.5626E-18, CF=3100., CM=5800., CS=7.38E+4, CPM=1.16, HFM=591., HTM=591., MDOTM=6280., TAU3=3., TAU4=3., TAU5=0.22151, C1=0.22151, C2=4.7712, ALPHA=5.64E-5, LAMDA=0.078, ALPHA1=3.1E-4, RSHIM=1.25E-4, TCDI=7.50, ERROR=0.05, DJMMY=0.0

PARAMETER


```

TSD = ( THID * (1.-K1) + IC00 * (1.+K1) ) * 0.5
PSD = EXP((ALDGC(TSD)-C2)/C1)
AD = LOAD0/100.

* NOSORT
*
* *
* * CALCULATE THE TURBINE LOAD CHANGE (LOADC) (%)
* * LOADC = LOAD3 - LOADF
*
* * SET THE DIRECTION OF TURBINE LOAD CHANGE
* * IF ( LOADC ) 5,13,15
* *   DLOAD = 1.
* *   GO TO 17
* *   DLOAD = 0.0
* *   RLOAD = 0.0
* *   GO TO 20
* *   DLOAD = -1.
*
* * CALCULATE THE TURBINE LOAD CHANGE TIME (TLOAD)
* * RLOAD = 0.005
* * TLOAD = ABS(LOADC)*2.0
*
* * PRINT INITIAL PLANT PARAMETERS
* * WRITE(6,5000) PWR0,TMR0,TH00,TC10,TSD,PSD,LOAD0,DSHIM,TSHIM,LOADF
*
* * CALCULATE DYNAMIC SECTION TRANSFER FUNCTION (TRANSF) COEFFICIENTS:
* * READ(TOR,C3) ANT JUTLET TEMPERATURE TRANSF COEFFICIENTS:
* *   C(1) = TAJ0/2. * (TAJ1/TAU2 + 1.) - TAU1
* *   C(2) = TAU0 * TAU1/2.
* *   C(3) = -1.
* *   D(1) = C(1) + 2.*TAJ1
* *   D(2) = C(2)
* *   D(3) = -C(3)
* *   C(1) = 0.0
* *   C(2) = GAMMA
* * XENDN TRANSFER FUNCTION COEFFICIENTS:
* *   C(3) = SIGMAF*U235
* *   C(4) = SIGMAX*FLUX0
* *   C(5) = SIGMAX*X0
* *   C(1) = (GAMMAX*C3 - C5)
* *   C(2) = (GAMDAI * (C3*(GAMMAI+GAMMAX) - C5)
* *   C(1) = L24 + LAMDAI + LAMDA
* *   C(2) = 1.
* *   C(3) = -LAMDAI*(C4+LAMDA)

```



```

** ** ** ** ** ** ** ** ** ** ** ** **  INSTANTANEOUS POWER DELIVERED TO TURBINE (%)
** ** ** **  PWRSGT  INSTANTANEOUS POWER DELIVERED TO THE TURBINE
** ** ** **  A      INSTANTANEOUS TURBINE THROTTLE VALVE SETTING
** ** ** **  (BTU/SEC)
** ** ** **  DYNAMIC
** ** **  REACTOR < INETICS
** ** **  RHO = RHOE + 3RHO + RHOX
** ** **  PWR = TRANSF(2,I,1,H,RHJ)
** ** **  PWR2 = PWR * 2.19E+3
** ** **  FLUX = PWR * 2.0E+12
** ** **  MODERATOR TEMPERATURE FEEDBACK
** ** **  RHOT = TMR * ALPHAT
** ** **  TMR = (THO+TCI)/2.
** ** **  XENON-135 FEEDBACK
** ** **  X = TRANSF(2,E,1,E,FLUX)
** ** **  RHOX = X * ALPHAX
** ** **  REACTOR AND STEAM PLANT ALGORITHM
** ** **  PRIMARY THERMAL LOOP
** ** **  THO1 = TRANSF(2,D,1,G,PWR2)
** ** **  THO2 = TRANSF(2,D,2,C,TCI)
** ** **  THO = THO1 - THO2
** ** **  TCIP = REALPL(0.0,TAU4,TCJ)
** ** **  TCI = REALPL(0.0,TAU6,TCIP)
** ** **  SECONDARY THERMAL LOOP
** ** **  THIP = REALPL(0.0,TAU3,THJ)
** ** **  THI = REALPL(0.0,TAU6,THIP)
** ** **  PWRSG = A * LCC.0
** ** **  LOAD = PWRSG
** ** **  PWRSGT = PWRSG * 2.19E+3
** ** **  PROCEDURE TCO = LCCP(PWRSGT,PWR2OT,TCOI,THI,TCI,ERROR,TAU5,K1)
** ** **  TCO = IMPL(TCOI,ERROR,FOFTCO)
** ** **  SGVARY = (THI+TCO)/2. - (PWRSGT-PWR2OT)/HTM
** ** **  TS = REALPL(0.0,TAU5,SGVARY)
** ** **  FOFTCO = (2.*TS - THI*(1.-K1))/(1.+K1)
** ** **  ENDPROCEDURE
** ** **  TST = TSO + TS
** ** **  PST = EXP((ALGG(TST)-C2/C1)
** ** **  ROD CONTROL
** ** **  RHOE1 = DSHIM * RSHIM * RAMP(TIME1)
** ** **  RHOE2 = DSHIM * RSHIM * RAMP(TIME2)
** ** **  RHOE = 3RHOE1 - RHOE2
** ** **  TURBINE THROTTLE CONTROL
** ** **  A1 = AO + DLJAD * RLJAD * RAMP(SLOAD)
** ** **  A2 = DLOAD * RLJAD * RAMP(TLJAD)
** ** **  A = A1 - A2

```



```

*      PROCEDURE POWER,XE135,TAV,TH,TC,SUR = RCS(SDFLAG,CFLAG1,CFLAG2, . . .
*      CFLAG3,TH=LAG,TFLAG,SFLAG1,SFLAG2,T1,T2,T,
*      SWRO,PWR,TMRO,TMR,THJO,THO,TCIO,TCI,XO,X,CFLAG4)
*
*      CHECK IF VALID INTEGRATION STEP PERFORMED
*      IF (KEEP .NE. 1) GO TO 115
*      CALCULATE INSTANTANEOUS PARAMETERS
*      POWER = PWRD + PWR
*      TAV = TMRO + TMR
*      TH = THO + THJ
*      TC = TCIO + TCI
*      XE135 = XO + X
*      CALCULATE REACTOR POWER STARTUP RATE (SJR)
*      PWRDGT = DERIV(PWRD,POWER)
*      SUR = PWRDGT * 0.06
*
*      REACTOR PROTECTION ALGORITHM
*      IF (TIME .LE. 0.05) GO TO 115
*      CHECK IF SCRAM IN PROGRESS
*      IF (SDFLAG .EQ. 1) GO TO 115
*      CHECK FOR HIGH POWER SCRAM 25
*      IF (CHECK FOR HIGH 138.) TEMPERATURE SCRAM CONDITION (TH>550F)
*      IF (TH .GE. 550.) GO TO 30
*      IF (CHECK IF ANY PREVIOUS CUTBACK HAS CLEARED
*      IF (CFLAG3 .EQ. 1) CFLAG1 = 0
*      IF (CHECK IF CUTBACK IN PROGRESS
*      IF (CFLAG1 .EQ. 1) GO TO 45
*      IF (CHECK FOR HIGH SJR (>1.74) CUTBACK CONDITION
*      IF (SUR .GE. 1.74) GO TO 60
*      CHECK FOR HIGH POWER CUTBACK CONDITION (POWER>114%)
*      IF (POWER .GE. 114.) GO TO 65
*      CHECK IF THOT ALARM EXISTS
*      IF (CHECK FLAG .EQ. 1) GO TO 85
*      CHECK IF THOT ALARM CONDITION EXIS (TH>544F)
*      IF (TH .LE. 544.) GO TO 90
*      IF (TH .LE. 544.) SET THFLAG
*      THFLAG = 1
*      PRINT NOTICE THOT ALARM
*      WRITE (6,6001) TH,TIME
*      GO TO 90
*
*      PRINT NOTICE HIGH POWER SCRAM
*      WRITE (6,6005) POWER,TIME
*      GO TO 40

```



```

* * * 30 PRINT NOTICE HIGH TEMPERATURE SCRAM
* * * 31 WRITE (6,6006) POWER,TH,TIME
* * * 32
* * * 33 SET SCRAM PARAMETERS:
* * * 34 SET RJD DIRECTION "IN"
* * * 35 DSHIM = -1.
* * * 36 SET CONTRJL RJD SCRAM INSERTION RATE
* * * 37 RSHIM = 0.19
* * * 38 SET TIME TO START SCRAM
* * * 39 TIME1 = TIME
* * * 40 SET TIME TO STOP SCRAM
* * * 41 TIME2 = TIME + 1.35
* * * 42 TRIP TURBINE THROTTLE
* * * 43 AO = 0.0
* * * 44 DLOAD = -1.
* * * 45 SLOAD = TIME + 1.
* * * 46 TLOAD = TIME
* * * 47 SET SDFLAG
* * * 48 SDFLAG = 1
* * * 49 GO TO 115
* * * 50
* * * 51 CHECK IF HIGH POWER CUTBACK CONDITION CLEARED
* * * 52 IF (CFLAG.EQ.1) GO TO 50
* * * 53 IF (POWER.GE.100.) GO TO 115
* * * 54 PRINT NOTICE HIGH POWER CUTBACK CONDITION CLEARED
* * * 55 WRITE (6,6007) POWER,TIME
* * * 56 GO TO 55
* * * 57
* * * 58 CHECK IF HIGH SUR CUTBACK CONDITION CLEARED
* * * 59 IF (SUR.GE.1.2) GO TO 115
* * * 60 PRINT NOTICE HIGH SUR CUTBACK CONDITION CLEARED
* * * 61 WRITE(6,6008) POWER,SUR,TIME
* * * 62
* * * 63 STOP CUTBACK
* * * 64 TIME1 = TIME
* * * 65 TIME2 = TIME
* * * 66 SET CFLAG2 (REOCCURRING CUTBACK WILL RESULT IN SCRAM)
* * * 67 CFLAG2 = 1
* * * 68 SET CFLAG 3 (ZERO CFLAG1)
* * * 69 CFLAG3 = 1
* * * 70 GO TO 115
* * * 71
* * * 72 PRINT NOTICE HIGH SUR CUTBACK CONDITION
* * * 73 WRITE (6,6009) POWER,SUR,TIME
* * * 74 CFLAG4 = 1
* * * 75 GO TO 70

```



```

LABEL      SIMULATION OF A PRESSURIZED WATER REACTOR
*
OUTPUT TIME,TAV(500.,550.),TH(500.,550.),TC(500.,550.)
LABEL      TEMPERATURE RESPONSE
PAGE HEIGHT = 5, WIDTH = 7
PAGE XYPLOT

*
OUTPUT TIME,POWER(0.,100.),LOAD(0.0,100.0)
LABEL      POWER AND LOAD RESPONSE
PAGE HEIGHT = 5, WIDTH = 7
PAGE XYPLOT

*
OUTPUT TIME,RHJ(-.01,.01),RHOE(-.01,.01),RHOT(-.01,.01)
LABEL      REACTIVITY RESPONSE
PAGE HEIGHT = 5, WIDTH = 7
PAGE XYPLOT

*
TIMER      FINTIM=600., OUTDEL=0.1, PRODEL=5.0
*
END
INPUT
ENDINPUT
RESET

*
TITLE      SIMULATION OF A PRESSURIZED WATER REACTOR
*
RANGE      RHO,RHOE,RHOT,RHOX,XE135,TAV,POWER,LOAD
PRINT      RHO,RHOE,RHOT,RHOX,XE135,TAV,POWER,LOAD
*
LABEL      SIMULATION OF A PRESSURIZED WATER REACTOR
*
OUTPUT TIME,POWER(0.,100.),LOAD(0.0,100.0)
LABEL      POWER AND LOAD RESPONSE
PAGE HEIGHT = 5, WIDTH = 7
PAGE XYPLOT

*
OUTPUT TIME,TAV(500.,550.),TH(500.,550.),TC(500.,550.)
LABEL      TEMPERATURE RESPONSE
PAGE HEIGHT = 5, WIDTH = 7
PAGE XYPLOT

*
OUTPUT TIME,XE135,TAV(500.,550.)
LABEL      XE AND AVG TEMPERATURE RESPONSE
PAGE HEIGHT = 5, WIDTH = 7
PAGE XYPLOT

```



```

* OUTPUT TIME,RHD(-.01,.01),RHDE(-.01,.01),RHOI(-.01,.01),RHOX(-.01,.01)
  LABEL {ACTIVITY RESPONSE
  PAGE XENON TRANSIENT
  HEIGHT = 5, WIDTH = 7
  PAGE XYPLOT
* TIMER FINTIM=5.4E+3,OUTDEL=1.0,PRDEL=60.0
*
  END
  INPUT
  ENDINPUT
  STOP
  ENDJOB

```


INTERFACE PROGRAM LISTING

```

&TYPEOUT ERROR
&TYPEOUT ON
VSET RDT T TEK-IB SYS-IB SSPLIB
GLOBAL LINE LV 80
CP SET LOT DESK INITA- DATA
FILEDEF IN TACT DESK INITAN >1 CORE CSMP PI INITIAL DATA PI
LOAD INE PLANT1 CSMP >1 PLANT1 CSMP PI
COMBINE PLANT2 CSMP >1 PLANT2 CSMP PI
COMBINE PLANT3 CSMP >1 PLANT3 CSMP PI
COMBINE PLANT CSMP PI
SUBMIT

```


LIST OF REFERENCES

1. Schultz, M. A., Control of Nuclear Reactors and Power Plants, 2nd ed., McGraw-Hill, 1961.
2. Bell, G. I. and Glasstone S., Nuclear Reactor Theory, 1st ed., Van Nostrand, 1963.
3. Radkowsky, A. (ed.), Naval Reactors Physics Handbook, vol. I, USAEC, 1964.
4. Henry, A. F., Nuclear Reactor Analysis, 1st ed., MIT, 1975.
5. Duderstadt, J. J. and Hamilton, L. J., Nuclear Reactor Analysis, 1st ed., Wiley, 1976.
6. The Shippingport Pressurized Water Reactor, personnel of the Naval Reactors Branch, AEC, and Westinghouse Electric, Addison-Wesley, 1958.

INITIAL DISTRIBUTION LIST

No. Copies

1. Defense Technical Information Center 2
Cameron Station
Alexandria, Virginia 22314
2. Library, Code 0142 2
Naval Postgraduate School
Monterey, California 93940
3. Department Chairman, Code 69 1
Department of Mechanical Engineering
Naval Postgraduate School
Monterey, California 93940
4. Professor Paul J. Marto, Code 69Mx 1
Department of Mechanical Engineering
Naval Postgraduate School
Monterey, California 93940
5. Professor Gilles Cantin, Code 69Ci 1
Department of Mechanical Engineering
Naval Postgraduate School
Monterey, California 93940
6. CDR C. J. Garritson, USN, Code 34 1
Naval Engineering Curricular Officer
Naval Postgraduate School
Monterey, California 93940
7. LCDR G. G. Heath, USN 1
Department Head Class 67
SWOSCOLCOM, Bldg. 446
Newport, Rhode Island 02840

Thesis
H41745 Heath
c.1

189507

An interactive code
for a pressurized water
reactor incorporating
temperature and Xenon
feedback.

Thesis
H41745 Heath
c.1

189507

An interactive code
for a pressurized water
reactor incorporating
temperature and Xenon
feedback.

An interactive code for a pressurized wa



3 2768 002 08673 8

DUDLEY KNOX LIBRARY